Appendix J

SHIELDING EVALUATION REPORT

Burns and Roe, Inc., performed the analysis of radiation levels occurring inside primary containment, assembled, edited, reviewed, and approved this technical report for Energy Northwest.

EDS Nuclear Incorporated performed the analysis of radiation levels occurring in the reactor building secondary containment under subcontract to Burns and Roe, Inc. Later revisions have been issued by Energy Northwest to incorporate plant changes.

Energy Northwest performed the analysis of radiation levels occurring in areas outside the reactor building secondary containment.

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SUMMARY

The Three Mile Island (TMI-2) accident has generated a concern that during an accident in which significant core damage occurs, the postaccident operations requiring the use of systems containing contaminated fluid may induce abnormally high radiation doses to safety-related equipment and components which make it difficult to operate the systems. The NRC initially addressed this concern with NUREG-0578 and NUREG-0737 and recommended a design review to evaluate the functional capability of safety-related equipment and radiation exposure to personnel during the postulated post-LOCA operations.

Radiation levels have been determined for all areas containing safety-related equipment, vital areas, and access routes which are required for the postulated post-LOCA operation.

Radiation levels determined for safety-related equipment inside primary containment. The analysis included the shadow shielding effect of installed equipment and the effect of iodine plateout were used to more accurately calculate the radiation levels inside containment.

Radiation levels were determined for safety-related equipment. The radiation source term leaking into secondary containment was reduced by the loss of halogens to plateout inside primary containment.

Radiation levels calculated for safety-related equipment outside secondary containment are reported in Table J.6-1.

Figures J.6-8 through J.6-18 identify the vital areas which require personnel access on either a continuous or infrequent basis during post-LOCA operations.

Safety-related equipment will either be qualified for the radiation level it functions in, or it will be relocated to a radiation zone it is qualified for, or it will be replaced with comparable equipment which is qualified for the particular radiation level that has been determined.

Vital areas and access routes were evaluated for post-LOCA operations and are reported in Table J.6-2 and Figures J.6-8 through J.6-18. All areas and access routes are in compliance with NUREG-0737.

ABSTRACT

This report presents a radiation shielding design review of the equipment and systems of the Energy Northwest Columbia Generating Station. The original report was prepared in September 1982. The equipment and systems are evaluated on the basis of a postulated accident which in addition to normal plant radiation levels during its 40-year life may contain highly radioactive fluids. This design review recommended by the NRC (NUREG-0578 and NUREG-0737) evaluates the functional capability of safety-related equipment and personnel radiation exposure during the postaccident operations.

This design review evaluates the postaccident radiation conditions for personnel located in vital areas (areas which require access or occupancy during the post-LOCA scenario) on either a continuous or infrequent basis.

The postulated loss-of-coolant accident (LOCA) scenarios and the operations of the safety-related systems were reviewed. Radioactive sources contained within each system were developed. Radiation levels were calculated at safety-related equipment locations, as well as at selected locations outside the reactor building to which access may be required for postaccident operations.

J.1 INTRODUCTION

This report presents a detailed description of the results and the review of plant shielding and radiation environmental conditions for equipment and systems which may be used in postaccident operations for Columbia Generating Station (CGS). The review was initiated in response to Section 2.1.6.b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendation," and to Part II.B.2 of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident."

The design review determined the postaccident radiation environmental conditions for equipment required for postaccident operations inside the primary containment, inside the secondary containment, and outside the secondary containment.

The 6-month total postaccident radiation dose rate as a function of time and the integrated dose were calculated at safety-related equipment locations inside the CGS reactor building, inside primary containment, and at selected locations (vital areas) outside the reactor building.

Section J.2 discusses the regulatory requirements on which this report is based and provides a description of the tasks performed for this shielding evaluation.

Section J.3 provides the systems review and source term assumptions used as input for the definition of the postaccident radiological environment.

Section J.4 discusses the work performed during this project relating to safety-related equipment located outside of the reactor building and the access and occupancy of vital areas. This consists of the calculation of dose rates outside the reactor building.

Section J.5 discusses the methods of calculation including the use of computer codes, identifying the parameters that have a significant effect on the radiation dose rates, and the dose rate and cumulative dose calculation procedure.

Section J.6 presents a summary of the results.

J.2 REQUIREMENTS

General Design Criterion 4 (10 CFR 50 Appendix A) requires that systems and components important to safety be designed to accommodate the environmental conditions associated with accidents. The Three Mile Island (TMI-2) accident has generated a concern that during an accident in which significant core damage occurs, the postaccident operations requiring the use of systems containing contaminated fluid may induce abnormally high radiation doses to safety-related equipment and components which may make it difficult to operate the systems. The NRC Lessons Learned Task Force initially addressed this concern in Section 2.1.6.b of NUREG-0578 (Reference J.7-1) and recommended a design review be performed on such systems so that the functional capability of safety-related equipment located in close proximity to the resulting high radiation field will not be unduly degraded.

Described in this section is a discussion of the current regulatory requirements and guidelines used.

J.2.1 SHIELDING EVALUATION REGULATORY REQUIREMENTS

NUREG-0578 Section 2.1.6.b requires that each licensee perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The scope of the review includes the following:

a. Identification of the locations of vital areas and safety-related equipment,

b. Evaluation of the radiation level at each location, and

c. Provision for adequate access to vital areas and assurances of postaccident equipment operation through design changes, increased permanent or temporary shielding, or postaccident procedural controls.

To perform this review, the NRC has provided guidance in the following documents ("documents of record"):

a. NUREG-0578, Section 2.1.6.b, Reference J.7-1,

b. NUREG-0588, Revision 1, Section 1.4, Reference J.7-2,

c. NUREG-0660, Section II.B.2, Reference J.7-3,

d. Clarification Letter to NUREG-0578, dated September 5, 1980, Section II.B.2, Reference J.7-4,

e. NUREG-0737, Section II.B.2, Reference J.7-5,

f. IE Bulletin No. 79-01B, Reference J.7-6, and

g.. IE Bulletin 79-01B, Supplement 2, dated September 30, 1980, Reference J.7-7.

The regulatory requirements in the above mentioned documents are summarized in the following sections.

J.2.1.1 Accident Analysis Requirements

The postaccident radiation environment should be based on the most severe design basis accidents (DBA) during or following which equipment must remain functional. This includes the consideration of the entire spectrum of loss-of-coolant accident (LOCA) events which can lead to a degraded core condition. These accident conditions include the following:

a. Loss-of-coolant accident events which completely depressurize the primary system, and

b. Loss-of-coolant accident events in which the primary system may not be depressurized.

J.2.1.2 Source Term Assumptions

The radioactive source terms for the postulated accident conditions as described in Section J.2.1.1 should be equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.7 and Standard Review Plan Section 15.6.5. The source term assumptions consistent with current licensing requirements used for equipment qualification and access evaluations are summarized as follows:

a. The fission product fractions assumed to be released from the fuel rods during a LOCA are the following:

Noble gases	100%
Halogens	50%
Remaining fission products	1%

For the analyses, 50% of the halogens and 1% of the solids were assumed to be diluted into the suppression pool and liquid carrying systems. The halogens were also assumed to be in the airborne source while iodines were assumed in the plateout source. Thus, some care is necessary in summing calculated doses to prevent double counting of the sources. The post-LOCA source contribution from liquid and plateout sources are analyzed separately and the worst dose is tabulated for that evaluation rather than the sum of both doses. Thus, double counting of the fission product fractions is eliminated where possible;

b. The above release is assumed to occur and be distributed instantaneously at the start of the accident. The plateout is assumed to occur over an effective time of 5 hr after the accident;

c. Until depressurized, liquid in the reactor coolant system (RCS) and other systems which are not isolated from the core and which contain the reactor coolant at the start of the LOCA contain 100% noble gases, 50% halogens, and 1% of the remaining fission products. These radioactive materials are mixed homogeneously in a volume no greater than the RCS liquid space;

d. Liquid in the suppression pool and any system not isolated from the core at the start of the LOCA, and containing only liquid from a depressurized source, is assumed to contain 50% halogens and 1% of the remaining fission products. These radioactive materials are diluted homogeneously in a volume no greater than the combined volumes of the suppression pool and the RCS liquid space;

e. The primary containment atmosphere and systems which are not isolated from the primary containment atmosphere at the start of the LOCA are assumed to contain at least 100% noble gases and 50% halogens initially. These radioactive materials are diluted homogeneously in a volume no greater than the combined volumes of the drywell and suppression pool air spaces; and

f. Primary containment plateout source term is obtained by allowing the airborne halogens released (50%) to plateout on primary containment surfaces in accordance with the guidelines presented in NUREG/CR-0009 until the airborne elemental iodine concentration is decreased by a factor of 200.

g. Until the reactor vessel is depressurized, gases in the steam lines and any other vapor-containing lines not isolated from the core at the start of the LOCA are assumed to contain at least 100% noble gases and 25% halogens. These are diluted uniformly in a volume no greater than the RCS steam space and adjoining unisolated steam lines.

J.2.1.3 Vital Area Access Requirements

As defined in NUREG-0737 (Reference J.7-5), a vital area is an area which will or may require occupancy to permit an operator to help in the mitigation of an accident or perform postaccident operations. The accident scenarios discussed in Section J.2.1.1 and the source term assumptions in Section J.2.1.2 are used for the evaluation of vital area access and occupancy. The total radiation exposure to personnel in vital areas should not be in excess of

5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident. For areas requiring continuous occupancy (e.g., the control room, onsite technical support center, etc.), the dose rate criteria limits the total radiation exposure to less than 15 mrem/hr (averaged over 30 days).

J.2.1.4 Systems Containing the Sources

Systems considered in the shielding review are those systems that could have the potential of containing a high level of radioactivity postaccident. For those systems connected directly to the RCS or to the primary containment atmosphere and not isolated at the start of the accident, the radioactivity is assumed to be instantaneously mixed within the unisolated parts of the system.

J.2.1.5 Safety-Related Equipment (C1E/SRM)

The safety-related (C1E/SRM) equipment list contains all equipment necessary to mitigate the consequences of an accident, bring the plant to a safe shutdown condition, and provide long-term cooling capability. This list includes equipment located inside as well as outside the primary containment.

J.2.2 SHIELDING EVALUATION TASK DESCRIPTION

The shielding evaluation tasks which have been completed to date are as follows:

a. Review all accident scenarios and accident conditions that could result in a limiting radiation environment for all the pieces of safety-related equipment on the C1E/SRM (safety-related) list that are located in the reactor building;

b. Identify systems and components that could potentially contain radioactive materials postaccident;

c. Generate source term assumptions based on regulatory requirements discussed in Section J.2.1;

d. Calculate accident radiation service conditions for the safety-related equipment located inside the reactor building;

e. Calculate gamma dose rates at selected locations outside the reactor building due to radioactive sources inside the reactor building;

f. Identify vital areas and equipment to evaluate the access to and occupancy of the vital areas in accordance with the requirements listed in Section J.2.1;

g. Conduct a primary containment analysis of LOCA events in which the RCS may not depressurize (or may repressurize) with a degraded core condition. The primary containment radiation environment was determined with the use of 100% noble gases, 50% halogens, and 1% of the remaining fission products for the period of time during which the activity is isolated to the RCS;

h. Calculate the radiation dose to safety-related equipment in the reactor building from post-LOCA airborne radiation and from normal piping sources inside primary containment streaming through the bioshield wall penetrations; and

i. The safety-related equipment list contains all equipment required to "mitigate" the consequences of an accident, bring the plant to a safe shutdown condition, and provide long-term cooling capability. The completeness of the safety-related equipment list has been verified.

J.2.3 SHIELDING EVALUATION ITEM DELETED FROM SHIELDING ANALYSIS CONSIDERATION

Columbia Generating Station has addressed all the issues needed to comply with the NUREG-660 II.B.2 position except as follows: Columbia Generating Station takes exception to the portion of the task that specifies that a review of "safety-related" equipment which may be degraded by radiation during postaccident operation be provided for a non-LOCA, high-energy line break source term. The pipe break/missile analysis described in Sections 3.5 and 3.6 addresses nonmechanistic pipe breaks inside and outside containment. These pipe breaks do not lead mechanistically to a radiation release due to fuel failures beyond those allowed in normal operation. Hence, the source term identified and applied outside containment is entirely hypothetical and would be a new design basis beyond the scope of current regulations.

J.3 ANALYTICAL METHODOLOGY

To develop the method used in the calculation of radiation doses, a review of all the postulated accident scenarios and system operations were performed. Source term assumptions were developed based on the results of accident analysis and system review, as well as the regulatory guidelines described in Section J.2.1. The systems and components inside the reactor building that have the potential of becoming contaminated during or following the accident were identified.

The following subsections describe these activities in greater detail. Section J.3.1 describes the accident scenario chosen for this analysis. Section J.3.2 identifies all the contaminated systems. Section J.3.3 describes the source term assumptions generated for each contaminated system. Section J.3.4 identifies the time period considered for this study.

J.3.1 ACCIDENT SCENARIO

The accident analyses consistent with FSAR Chapter 15 for small- and large-break loss-of-coolant accidents (LOCAs) were considered. The entire spectrum of LOCA conditions that could result in a degraded core configuration was reviewed and it was concluded that there is no single accident scenario that could result in a limiting radiation environment for all the safety-related equipment located in the reactor building. Therefore, the accident scenario chosen here is based on a nonmechanistic LOCA in which core damage is experienced at the beginning of the accident. Primary containment isolation is assumed to be achieved prior to radioactivity transport.

A review of the postaccident operation of the C1E/SRM (safety-related) systems was conducted. The result of this review indicated that the worst-case accident for the steam supply system (highest source term) was the pressurized reactor coolant system (RCS). For the liquid systems [the emergency core cooling system (ECCS), the residual heat removal (RHR), and the reactor core isolation cooling (RCIC) systems], as well as the primary containment atmosphere and primary containment atmosphere control (CAC) system, the worst-case accident is the depressurized reactor coolant system with the post-LOCA core release functions dispersed within the primary containment.

J.3.2 CONTAMINATED SYSTEMS

To perform the radiation dose calculations, it was necessary to identify the systems which would or could contain highly radioactive materials during the postaccident period. Systems required to operate during the postaccident period are as follows:

a. Systems necessary to mitigate the consequences of a large- or small-break LOCA,

b. Portions of systems that are in communication with systems containing radioactive liquids or gases, and

c. Defined by the NRC as being required, such as the gaseous radwaste system (see Section J.3.2.3).

J.3.2.1 Systems Included for Primary Containment Analysis

The following systems were considered:

a. High-pressure core spray (HPCS),

b. Low-pressure core spray (LPCS),

c. RHR,

d. RCIC,

e. Floor drains and equipment drains (FDR-EDR),

f. Reactor water cleanup (RWCU),

g. Main steam (MS),

h. Reactor recirculation (RRC),

i. Sample lines (PSR),

j. Automatic depressurization system (ADS), and

k. Low-pressure coolant injection (LPCI) function of the RHR system after depressurization.

J.3.2.2 Systems Included for Secondary Containment Analysis

The following systems were considered:

a. RCIC,
b. RHR,
c. LPCI,
d. LPCS,
e. HPCS,
f. MS, up to second isolation valve,

g. MS line isolation valve-leakage control system (MSIV-LCS),
h. Primary containment,
i. Secondary containment atmosphere, and
j. Standby gas treatment (SGT).

The following systems were also considered due to their potential to affect isolation valves or extend the primary containment source terms into secondary containment.

a. Containment atmosphere monitoring (CMS),

b. Containment supply purge (CSP),

c. Containment exhaust purge (CEP),

d. Blank penetrations,

e. Personnel access doors into the wetwell and drywell,

f. Instrumentation penetrations, and

g. All post-LOCA inboard and outboard isolation valves and their connected piping sources.

J.3.2.3 Systems Excluded

All systems required to mitigate the consequences of an accident have been included. Of those systems recommended for consideration in regulatory documents, one system (gaseous radwaste) has been excluded.

The gaseous radwaste is isolated by the primary containment and reactor vessel isolation control system and will not receive contaminated gas unless operation is manually initiated. The Columbia Generating Station (CGS) operating and accident procedures do not take credit for nor anticipate using this system. Since CGS philosophy is based on containment of the core releases within the primary containment, this system will not be required and was, therefore, excluded from consideration.

J.3.3 SOURCE TERM ASSUMPTIONS

Regulatory requirements specify that source terms equivalent to those recommended in Regulatory Guides 1.3 and 1.7 and Standard Review Plan Section 15.6.5 be used in the LOCA accident analysis. Additional guidance is given in NUREG-0588 (Reference J.7-2) and NUREG-0737 (Reference J.7-5) and is documented in Section J.2.1. Source term assumptions were generated based on the review of the operation of the safety systems. Because a nonmechanistic LOCA scenario was chosen for this analysis, the worst contaminated situation for the fluid contained within each system was conservatively assumed. Tables J.3-1, J.3-2, and J.3-3 list the assumptions involved in the distribution of fission products used in this analysis. These assumptions are consistent with the regulatory requirements discussed in Section J.2.1.

A review of the operation of each of the systems discussed in Section J.3.2 was also conducted. This review identified the source of contaminated fluid contained within each system postaccident. Using the source term assumptions discussed in Tables J.3-1, J.3-2, and J.3-3, together with the results of this system review, the limiting source term (activity divided by dilution factor) was determined for each system. Table J.3-4 is a summary of the system operations and source term assumptions developed for each contaminated system identified in Section J.3.2.

J.3.4 TIME PERIOD CONSIDERED FOR STUDY

All systems were conservatively assumed to become contaminated at the start of the accident and remain contaminated until the integrated radiation dose reached it asymptotic value. It was noted that the integrated dose becomes nearly asymptotic to a constant value beyond about 6 months. Therefore, 6 months is the time period chosen for accident dose qualification in this report.

Table J.3-1

Distribution of Fission Products in the Worst
Post-Loss-of-Coolant Accident Situation for Areas
Inside Containment Depressurized Reactor Coolant System

	Primary Containment [a] Air and Steam Space		Suppression Pool and Reactor Coolant System Water Volume	
Fission Products	Fraction [b]	Dilution Volume [c]	Fraction[b]	Dilution Volume [c]
Noble gases	100%	Drywell air plus	0%	Suppression pool water and RCS water volume
Halogens	50% [d,e]	Suppression pool	50%	
Particulates	0%	Air	1%	

[a] A uniform distribution between drywell and suppression pool atmosphere has been assumed.

[b] Expressed in percentage (%) of total core inventory at end-of-life conditions (1000 days at 3556 MWt).

[c] Represents the total volume in which the fraction of core fission products is assumed to be homogeneously mixed.

[d] In calculating the radiation dose at a particular location, it is not necessary to assume that all source distribution assumptions are conservative simultaneously. Instead, a set of mutually compatible assumptions will be used which gives the maximum dose for the location being considered. The post-LOCA source contributors are used to calculate independent doses for each contributor. The worst dose is tabulated for that system rather than the sum of all contributors (i.e., 50% halogens airborne and 50% halogens in the water). Thus double counting of the fission product fractions is eliminated.

[e] First order iodine plateout occurs during the first 5-6 hr of the post-LOCA time frame when the elemental halogen concentration is reduced by a factor of 200. This methodology is in accordance with NUREG/CR-0009. Of the halogens released, 95.5% is available for plateout. Virtually all of the available halogens plateout within the initial 5 hr after the accident (0.5% remain airborne).

Table J.3-2

Distribution of Fission Products in the Worst
Post-Loss-of-Coolant Accident Situation for Areas
Inside Containment Pressurized Reactor Coolant System[a]

	Drywell Air Space[a]	Suppression Pool Water Volume and Air Space[a]	Reactor Coolant System Water Volume[a]		Reactor Coolant System Steam Space[a]	
Fission products	Fraction[b]	Fraction[b]	Fraction[b]	Dilution Volume[c]	Fraction[b]	Dilution Volume[c]
Noble gases	0%	0%	100%[d]	RCS water volume[e]	100%[e]	Normal RCS steam space[f]
Halogens	0%	0%	50%[g]		25%	
Particulates	0%	0%	1%		0%	

[a] The reactor coolant system will remain pressurized for a short period of time (17 hr) and then will be depressurized.

[b] Expressed in percentage (%) of total core inventory at end-of-life conditions (1000 days at 3556 MWt).

[c] Represents the total volume in which the fraction of core fission products is assumed to be homogeneously mixed.

[d] The 100% of noble gases, present during the 17 hr of the pressurized RCS during a LOCA, are homogeneously mixed in the water and steam dilution volumes identified.

[e] The dilution volume is the RCS water volume plus the RWCU lines up to the isolation valves, RHR lines to the isolation valves, and the RRC lines during the 17 hr of the pressurized RCS scenario.

[f] The dilution volume is the normal RCS steam space plus the MS lines up to the isolation valves during the 17 hr of the pressurized RCS scenario.

[g] In calculating the radiation dose at a particular location, it is not necessary to assume that all source distribution assumptions are conservative simultaneously. Instead, a set of mutually compatible assumptions will be used which gives the maximum dose for the location being considered. The post-LOCA source contributors are used to calculate independent doses for each contributor. The worst dose is tabulated for that system rather than the sum of all contributors (i.e., 50% halogens airborne and 50% halogens in the water). Thus double counting of the fission product fractions is eliminated.

Table J.3-3

Distribution of Fission Products in the Worst Post-Loss-of-Coolant Accident Situation for Areas Outside Containment

	Primary Containment Air Space		Suppression Pool Water Volume		Reactor Coolant System Steam Space[a]		Reactor Coolant System Water Volume[a]	
Fission Products	Fraction[b]	Dilution Volume[c]	Fraction[b]	Dilution Volume[c]	Fraction[b]	Dilution Volume[c]	Fraction[b]	Dilution Volume[c]
Noble gases	100%	Drywell	0%	Suppression pool water plus RCS water	100%	Normal	100%	RCS
Halogens	50%[d]	Air plus	50%[e]		25%	RCS	50%	Water
Particulates	0%	Suppression pool air	1%		0%	Steam space	1%	Volume

[a] Based on pressurized reactor coolant system.

[b] Expressed in percentage (%) of total core at end-of-life conditions (1000 days at 3556 MWt).

[c] Represents the total volume in which the fraction of core fission products is assumed to be homogeneously mixed.

[d] 95% of the halogens released from the core are assumed to plateout within approximately 5 hr as allowed by NUREG/CR-0009. The plateout dose was considered in the total calculation of radiation dose to equipment inside primary containment.

[e] In calculating the radiation dose at a particular location, it is not necessary to assume that all source distribution assumptions are simultaneously conservative. Instead, a set of mutually compatible assumptions will be used which gives the maximum dose for the location being considered. The post-LOCA source contributions are used to calculate independent doses for each contributor. The worst dose is tabulated for that system rather than the sum of total contributors (i.e., 50% halogens airborne and 50% halogens in the water). Thus double counting of the fission of product fractions is eliminated.

Table J.3-4

System Operation and Source Term Assumptions

System	Operation Postaccident	Contaminated Space	Source Term Assumptions
HPCS	Suction from condensate storage tank and/or suppression pool and discharge to the reactor vessel.	Suppression pool	(1)
LPCS	Suction from suppression pool and discharge to the reactor vessel.	Suppression pool	(1)
LPCI	Suction from suppression pool and discharge to the core.	Suppression pool	(1)
(6) RCIC steam system	Steam bleed-off from reactor steam space is used to drive the RCIC turbine, and exhausts into the suppression pool.	RCS steam space	(2)
RCIC liquid system	Suction from condensate storage tank or suppression pool and discharge to the reactor vessel.	Suppression pool	(1)
RHR system	(1) Shutdown cooling mode - suction from reactor recirculation system suction line and discharge into the reactor recirculation discharge line.	RCS liquid space	Note a
	(2) Alternate shutdown cooling mode - suction from suppression pool and discharge to core recirculates and cools the water in the suppression pool.	Suppression pool	(1) Note b
	(3) Containment spray cooling mode - suction from suppression pool and discharge into the drywell and suppression pool.	Suppression pool	(1)
	(4) Reactor steam condensing mode.	System mode deleted from plant	

Table J.3-4

System Operation and Source Term Assumptions (Continued)

System	Operation Postaccident	Contaminated Space	Source Term Assumptions
Main steam supply (MS)	Stagnant steam from the reactor vessel terminates at the second MSIV.	RCS steam space	(2)
MSIV-LCS (MSLC)	Steam bleed-off from main steam line, diluted, and discharged into the SGTS.	RCS steam space	(2) Note c
SGT filters (SGTS)	Process the halogens from primary containment leakage and MSIV-LCS.	Primary containment and secondary containment atmosphere	(3)
Primary containment (PCN)	Primary containment is isolated postaccident.	Primary containment atmosphere	(4)
Suppression pool	The primary function of the suppression pool is to contain and condense the blowdown from the RCS postaccident.	Suppression pool liquid	(1)
Secondary containment (SCN)	The primary function of the secondary containment is to contain all the leakage from the primary containment postaccident.	Primary containment atmosphere	(5)
Sample lines	Actuated to obtain primary containment atmosphere samples per NUREG-0737 (Reference J.7-5).	Primary containment atmosphere	(2)
Sample lines	Actuated to obtain liquid samples per NUREG-0737.	RCS liquid space	(1)
Reactor water cleanup (RWCU)	Reactor water cleanup system isolated during post-LOCA. Liquid up to the second isolation valve is considered contaminated.	RCS liquid	(1)
Reactor recirculation (RRC)	Suction from RRC system suction line and discharge into the reactor recirculation discharge line.	RRC liquid; RCS liquid	(1)
Floor drains and equipment drains (FDR/EDR)	Liquid from ruptured pipes or leaky seals discharged into the suppression pool.	RCS liquid	(1)

Table J.3-4

System Operation and Source Term Assumptions (Continued)

System	Operation Postaccident	Contaminated Space	Source Term Assumptions
Automatic depressurization system (ADS)	Automatic or manual depressurization of the reactor vessel by blowdown of the RCS into the suppression pool.	RCS steam	(2)
Automatic depressurization system (ADS)	Alternate shutdown cooling mode with reflood of reactor vessel and discharge into suppression pool.	Suppression pool	(1)
Containment monitoring system (CMS)	Continues to monitor primary containment atmosphere conditions.	Isolation of primary containment into secondary containment	(4)
Containment supply purge (CSP)	Isolated - no action required.	Isolation of primary containment into secondary containment	(4)
Containment exhaust purge (CEP)	Isolated - no action required.	Isolation of primary containment into secondary containment	(4)
Blank penetrations	None	Isolation of primary containment into secondary containment	(4)
Personnel access doors to primary containment	None	Isolation of primary containment into secondary containment	(4)
Instrumentation penetrations	None	Isolation of primary containment into secondary containment	(4)
All post-LOCA inboard and outboard isolation valves	As defined per Columbia Generating Station system requirements post-LOCA	Isolation valves and their connected piping which extends into secondary containment	Note d

<u>Source Term Assumptions</u>

(1) 50% halogens and 1% solid fission products diluted with suppression pool water plus RCS water.

(2) 100% noble gases and 25% halogens diluted with the RCS steam space.

Table J.3-4

System Operation and Source Term Assumptions (Continued)

(3) 50% halogens leaked from the primary containment is assumed to be deposited in the SGT filters at the rate of 0.67% per day. See Section J.5.3.3.1 for justification. 100% noble gases pass through also but are not absorbed.

(4) 100% noble gases and 50% halogens diluted with the primary containment air space. First order iodine plateout (0-95% elemental iodine) inside primary containment was considered.

(5) Assumptions involved in the calculation of source terms for secondary containment atmosphere are discussed in Section J.5.3.2.1.

(6) Based on a pressurized reactor coolant system.

[a] According to accident mitigation procedures, this mode of operation is not used after a degraded core condition is identified.

[b] Full discussion of source term assumptions for alternate shutdown cooling are presented in Section J.5.3.3.1.

[c] For the portion of system after the distribution header, credit is taken for dilution by clean air. See Section J.5.3.3.1 for justification.

[d] For all isolated systems the source term for the isolation valves will be primary containment atmosphere unless the penetration is filled with water that remains during the post-LOCA scenario. All penetrations and their associated isolation valves which contain a flowing fluid during post-LOCA operations are analyzed with the post-LOCA source term of that flowing fluid.

J.4 ACCESS AND OCCUPANCY OF VITAL AREAS

NUREG-0578 initiated the requirement for a design review to identify the location of vital areas in which personnel occupancy may be unduly limited by the radiation fields during postaccident operations. It required that each licensee provide adequate access to vital areas through design changes, increased permanent or temporary shielding, or postaccident procedural controls. NUREG-0737 further makes the point that the purpose of this design review is to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident.

This shielding evaluation includes the calculation of gamma dose rates at selected locations outside the reactor building due to radioactive sources inside. The radioactive source terms obtained from ORIGEN computer calculations coupled with recommendations from Regulatory Guide 1.109 were the basis for the assumptions used in evaluating vital areas and access routes outside the reactor building.

J.4.1 DOSE RATES OUTSIDE THE REACTOR BUILDING

An analysis was conducted to determine the dose rates at selected locations outside the reactor building for personnel access purposes. The radiation level in the various areas outside the reactor building is defined by the following three radioactive sources:

a. Direct gamma ray dose from radioactive piping located inside the reactor building and attenuated through the walls of the reactor building,

b. Gamma shine dose from airborne activity inside the reactor building, and

c. Gamma dose from airborne activity outside the reactor building.

Radiation levels outside the reactor building were determined by the zone dose method as discussed in Section J.5.4. Representative zones were chosen at selected locations outside the reactor building such as ground level outside the railroad bay, sampling room, etc. The worst point in a zone was chosen to be the point directly outside the reactor building wall, at a height of 6 ft above floor elevation, at a lateral point determined by inspection to receive the highest dose along that wall.

The zones outside the reactor building are indicated by the letters Y and Z in the various elevations. The shine dose contribution to areas outside the reactor building (Zones Y and Z) were included in the dose calculations shown in Figures J.6-11 through J.6-18.

Attachment J.H presents the methodology used to calculate the radiation doses for the various vital areas.

J.4.2 VITAL AREAS AND ACCESS ROUTES OUTSIDE THE REACTOR BUILDING

Radiation calculated for the access routes were based on the assumption that no individual would be in an access route longer than 30 minutes for the first 8 hr after the postulated LOCA before reaching the vital area of interest.

The assumption was also made that no individual would occupy an infrequent occupied vital area longer than 30 minutes for the first 8 hr after the postulated LOCA.

All integrated radiation doses calculated for time spent in the access routes and vital areas were less than the guidelines presented in NUREG-0737.

J.4.3 VITAL AREAS AND ACCESS ROUTES INSIDE THE REACTOR BUILDING

Analysis has been completed to take credit for a vital area in the reactor building railroad bay and on the west side of the 522-ft el. The analysis of reactor building zones is discussed in Section J.5.3. The access route to the reactor building is discussed in Sections J.4.1 and J.4.2. See Section J.6.3 for a description of the access to the 522-ft el. of the reactor building.

J.5 METHODS

Due to the large number of C1E/SRM components in primary containment, it was decided to calculate the worst point dose from each of the major sources in the drywell and wetwell, and then sum the doses for a conservative estimate of the total integrated dose.

The secondary containment radiation dose assessment portion of the shielding evaluation was initiated by dividing the reactor building into radiation zones. Because of the large number of radioactive piping and safety-related equipment in the building, the division of the various regions of the secondary containment into radiation zones permits a precise, detailed calculation of the total integrated dose at the "worst target" location. The methods for performing the calculations are discussed in detail in the following sections.

The radiation dose assessment of safety-related equipment outside of the reactor building was done by calculating the radiation dose of each vital area where safety-related equipment was located. The assumptions and methodology used to perform these calculations are discussed in detail in the following sections and in Attachment J.H.

J.5.1 THE USE OF COMPUTER CODES

The two computer codes used in the primary containment shielding evaluation were ORIGEN2 and QAD-CG. Descriptions of the two codes are found in References J.7-8, J.7-9, J.7-10, and J.7-17. ORIGEN2 was used to compute the radioactive source terms (inside containment) used by QAD-CG to calculate the radiation doses from piping and various pieces of equipment.

The three computer codes used in the original secondary containment radiation shielding review were ORIGEN, SCAP-BR, and QAD-P5A. Descriptions of the codes are found in References J.7-10, J.7-11, and J.7-18. ORIGEN computes the radioactive source terms used by QAD-P5A to compute the radiation from piping and other source configurations to pieces of equipment. SCAP-BR computes the radiation dose contribution to safety-related equipment in the reactor building from primary containment airborne radiation streaming through the bioshield wall penetrations.

ORIGEN and ORIGEN2 are fission product source term codes which solve the equations of radioactive growth and decay for large numbers of isotopes. The codes have been used to calculate the radioactivity of fission products and fuel materials that were assumed to be released from the reactor core during the postulated loss-of-coolant accident (LOCA) to become the primary containment source terms for the dose rate calculations. SCAP-BR is similar to QAD-CG with the added capability to determine the radiation dose contribution due to scattering.

J.5.2 SOURCE TERM DEVELOPMENT FOR PRIMARY CONTAINMENT

The radiation level at any given location inside the primary containment of Columbia Generating Station (CGS) following the postulated LOCA such as that described in Section J.3.1 is determined from the following major source contributors:

a. Gamma ray dose from airborne radioactive sources suspended in the drywell and wetwell inside primary containment (airborne gamma dose),

b. Gamma ray dose from piping and/or equipment containing contaminated fluids which are recirculated inside primary containment (direct gamma dose),

c. Gamma and beta ray dose from iodines plated out inside primary containment (iodine plateout), and

d. Beta ray dose emitted by airborne radioactive sources suspended in the drywell and wetwell inside primary containment (airborne beta dose).

The initial phase of this analysis was concerned with the determination of radioactive source terms for the liquids and gases inside primary containment. The ORIGEN2 computer code was used for this calculation. The fission product inventory at the end of fuel life (1000 days irradiation at a power level of 3556 MWt) was assumed to be available for release immediately following the accident. The release fractions and resulting concentrations of noble gases, halogens, and other fission products in the gaseous and liquid fluids were computed. A detailed description of the analysis including the assumptions used is provided in Attachment J.F.

J.5.3 SOURCE TERM DEVELOPMENT FOR SECONDARY CONTAINMENT

The radiation level at a given location inside the secondary containment of CGS following an accident such as that described in Section J.3.1 is defined by the following major source contributors:

a. Gamma ray dose from airborne radioactive sources inside secondary containment (airborne gamma dose),

b. Gamma ray dose from radioactive sources suspended in the drywell and the wetwell inside primary containment (containment shine dose),

c. Gamma ray dose from piping and/or equipment containing contaminated fluids which are recirculated inside the reactor building (direct gamma dose),

d. Beta ray dose emitted by airborne radioactive sources inside secondary containment (airborne beta dose), and

e. Gamma ray dose from liquid piping and airborne radioactive sources inside primary containment which stream through bioshield wall penetrations into secondary containment (bioshield penetration streaming dose).

The initial phase of this analysis was concerned with the definition of radioactive source terms for the liquid and gas containing systems. The ORIGEN computer code was used for this calculation. The fission products at the end of fuel life (1000 days irradiation at a power level of 3556 MWt) were assumed to be available for release immediately following the accident. The released fractions of noble gases, halogens, and other fission products to the gaseous and liquid sources were computed. Subsequent fission product depletion and daughter product generation were then calculated for 20 time periods, covering a total period of 1 year. A detailed description of the analysis, including the assumptions used, as well as results of the source terms, is found in Attachment J.B and Reference J.7-12.

J.5.3.1 Parametric Studies for Direct Piping Dose in Secondary Containment

The purpose of the parametric study was to identify the parameters which have a significant effect on the radiation dose rates inside secondary containment. The computer code QAD-P5A was used to develop a correlation scheme for the significant parameters such that a simplified procedure for calculating radiation dose rates for complex source and receptor geometries can be developed. The dose rate at a target distance of 8 ft radially outwards from the centerline of an 8-in. schedule 40 pipe, infinitely long (standard pipe), was first calculated and defined as the standard dose rate. The results of this parametric study were then correlated as a set of correction factors to the standard dose rate. A simplified procedure was developed to calculate the dose rates and cumulate doses for complicated source-target configurations by using these correction factors. The development of these correction factors and the result of the parametric study inside secondary containment is discussed in Attachment J.B.

J.5.3.2 Dose Rate and Cumulative Dose Calculation Procedure

The results of the source term calculations and those of the parametric study were used to generate and cumulate doses for complicated source target configurations inside secondary containment. The following steps were taken to define the radiation service conditions for the pieces of safety-related equipment:

a. Based on the accident scenarios, contaminated systems, and assumptions defined in Section J.3, the radioactive source terms for liquid-containing and gas-containing systems were developed;

b. Radiation zones were selected and the radiation zone boundaries were carefully defined based on shield wall locations, contaminated piping locations, and locations of safety-related C1E/SRM equipment;

c. The radiation environment in each secondary containment zone (zone dose) was calculated (see Attachment J.B for the procedure). A zone dose is the radiation dose (gamma) that bounds the magnitude of dose received by all the pieces of safety-related C1E/SRM equipment located within that zone;

d. The zone dose as calculated in step c was used, as a first cut, to qualify all the pieces of safety-related C1E/SRM equipment located within that zone; and

e. For the pieces of safety-related C1E/SRM equipment that could not be qualified for the conservative radiation environment calculated in step c, the integrated dose for that piece of equipment was redefined based on a more realistic and refined approach.

J.5.3.2.1 Calculation of Airborne Gamma Doses Inside Secondary Containment

The time-dependent post-LOCA activity levels as calculated by the ORIGEN computer code were used as input for the calculation of the airborne gamma dose rates and integrated doses inside the cubicles in the secondary containment. The assumptions used in this analysis are as follows:

a. Activity that leaks into the secondary containment is homogeneously mixed with the secondary containment atmosphere prior to its removal from the atmosphere through the standby gas treatment system (SGTS);

b. The SGTS flow rate of 2430 scfm was assumed to be the flow rate of the effluent air. This is equivalent to one reactor building air change per day;

c. Air that leaks out of the primary containment flows directly and totally into the secondary containment. Bypass leakage was not considered;

d. Geometric factors were used to convert the semi-infinite cloud gamma dose to a finite gamma dose; and

e. Primary containment leakage rate of 0.5 wt %/day was considered.

Justifications of the above assumptions are stated in Attachment J.B. The equations that were used for the gamma dose calculations are described in Attachment J.B. Primary containment airborne beta dose results are discussed in Attachment J.G.

J.5.3.2.2 Procedure for the Calculation of Radiation Zone Dose in Secondary Containment

As discussed previously, the gamma radiation level at a given location inside the secondary containment of CGS following a LOCA is determined for four types of radioactive source distributions:

a. Fission products suspended in the atmosphere of the secondary containment (airborne gamma dose),

b. Gamma irradiation from the primary containment (shine dose),

c. Direct gamma irradiation from the radioactive fluid contained inside recirculating pipes (direct dose), and

d. Gamma ray dose from liquid piping and airborne radioactive sources inside primary containment which stream through bioshield wall penetrations into secondary containment (bioshield penetration streaming dose).

The dose contributed by each of these sources is determined by the location of the equipment, the time-dependent distribution of the source, and the effects of shielding.

A step-by-step procedure for calculating radioactive zone doses is shown in Attachment J.C. The methods presented in that procedure make it possible to calculate the worst case gamma dose from the above-mentioned source contributors inside radiation zones of the secondary containment. In general, this procedure for determining zone doses consists of a correction factor method for calculating direct dose rates.

As discussed in Attachment J.B, the correction factor method for calculating dose rates provides a convenient and fairly precise way of determining direct dose rates due to generic pipe segments. For radioactive fluid contained within components of geometry other than generic pipe segments, such as residual heat removal (RHR) heat exchangers, SGTS filters, hydrogen recombiners, etc., special QAD-P5A computer modeling was performed to calculate the gamma dose contribution due to those systems. A brief description of the guidelines used in modeling special components is found in Attachment J.B.

An evaluation of beta dose is necessary for qualification of safety-related equipment that is beta sensitive and not adequately protected against beta radiation. The beta dose analysis for secondary containment is presented in Section J.5.5. Beta dose is discussed in more detail in Attachment J.D as related to secondary containment radiation contributors.

J.5.3.3 Calculation of Radiation Doses Due to Special Systems and Components Inside Secondary Containment

As discussed in Attachments J.B and J.C, the correction factor method for calculating gamma dose rates and integrated doses is involved with the application of the dose correction factors (pipe diameter, pipe length, and radial distance correction factors) to a standard dose rate curve. A standard dose is defined as the gamma radiation measured at a target distance of 8 ft and emitted by radioactive sources contained within the suppression pool liquid and recirculated within infinitely long 8-in. schedule 40 piping. The systems that contain such radioactive fluids are the reactor coolant system, high-pressure core spray, low-pressure core spray, and residual heat removal systems. Other systems which contain fluids of different source terms and dilutions are considered special sources. The systems that need to be considered for special sources are the following:

a. Standby gas treatment system filters,
b. Main steam system, and
c. Main steam isolation valve leakage control system (MSIV-LCS).

J.5.3.3.1 Source Term Assumptions in Secondary Containment

The assumptions for the calculations of source terms inside secondary containment for special source systems are listed as follows:

Standby Gas Treatment System Filters

a. The SGTS filters will be loaded by halogens at the rate of 0.67% primary containment free volume per day. This consists of 0.5% per day of primary containment leakage and 0.17% per day of leakage due to the MSIV-LCS system. No holdup of this activity in the secondary containment is assumed;

b. The released halogen fraction is 50% of the core halogen inventory. This halogen fraction is assumed to be composed of 95.5% elemental, 2% organic, and 2.5% particulate halogens; and

c. The particulate halogens are assumed to be homogeneously distributed within the prefilters and the particulate filters, while the elemental and organic halogens are assumed to be homogeneously distributed within the charcoal filters.

Assumption a is consistent with the assumptions used in the accident analysis (Reference J.7-13 and Section J.3.1).

Assumption b is the NRC recommended assumption for the distribution of halogen inventory (Reference J.7-14).

Assumption c is necessary because the time-dependent distribution of activity within a filter is unknown. The homogeneous assumption, therefore, is considered appropriate and conservative for zone dose assessment.

Containment Atmosphere Control System

The function of the CAC system was to process the primary containment atmosphere to remove oxygen after a LOCA accident. Therefore, this system was assumed to be filled with gaseous source containing 2.5% halogens and 100% noble gases diluted with the primary containment free volume, although it is now deactivated.

Main Steam System

The main steam lines are located inside and outside the primary containment; they include the main steam lines in the steam tunnel and the RCIC turbine supply and exhaust lines. The radioactive source term for this system is assumed to be composed of 100% noble gases and 25% halogens, distributed throughout the reactor coolant system (RCS) steam space.

Alternate (Suppression Pool) Shutdown Cooling

To prevent failure of the RHR pumps due to excessive radiation exposure, the alternate shutdown cooling mode is the only allowable mode for shutdown cooling once a degraded core condition has been identified.

A small pipe-break accident will take approximately 6 hr to depressurize from 1000 psi to 150 psi through automatic depressurization system (ADS) valve actuations. Once a degraded core is identified and the reactor is sufficiently depressurized, within 17 hr after the accident, the ADS valves actuation will be maintained to dilute the primary coolant source concentration with the suppression pool since the alternate shutdown cooling mode will be used for decay heat removal.

For the large pipe-break accident the primary coolant source concentration will be diluted by the water in the suppression pool due to blowdown of the vessel through the large break or automatic actuation of the ADS valves. Once the vessel has been depressurized the water level in the vessel will be maintained with the emergency core cooling system while decay heat is removed by suppression pool cooling.

Thus, in all degraded core scenarios the primary coolant is diluted with the suppression pool prior to initiating the suppression pool shutdown cooling mode.

Main Steam Isolation Valve Leakage Control System

The MSIV-LCS system is a vacuum-type system which collects leakage between and downstream of the closed isolation valves and then releases it to the atmosphere through the SGTS. Leakage through the valve stems (maximum leakage of 11.5 scfh as described in Reference J.7-15) is directed to a distribution header or low-pressure manifold where clean air is brought in to dilute the contaminated steam before exhausting to the SGTS filter unit at a rated flow rate of 50 scfm. Thus the source term in the portion of piping system before the distribution header is conservatively assumed to be the same as that of the main steam system. For the portion of the piping after the header, credit is taken for the dilution by the clean air. This assumption is consistent with that recommended in Reference J.7-16.

J.5.3.3.2 Secondary Containment Analysis Method

The correction factor method is used for the calculation of the direct dose contribution due to the piping systems described in Section J.5.3.3, with the exception of the SGTS filter system. A description of the analysis of the SGTS filter is documented in Attachment J.D. Generic piping dose rate and integrated dose (dose at a target distance of 8 ft away from the centerline of an infinitely long 8-in. schedule 40 pipe) for each system were developed using the source term assumptions discussed in Section J.5.3.1 and are shown in Attachment J.B. Parametric studies were also performed to investigate the variation of dose rates due to pipe diameter, pipe length, and target distance for pipe segments containing source terms. The gaseous source term correction factors derived as a result of this parametric study (described in Attachment J.B), together with the generic dose rate curves generated for each system, were used to calculate the direct gamma dose contribution on a target.

J.5.3.3.3 Calculation of Radiation Doses Inside Secondary Containment on Generic Mechanical Equipment

Table J.5-1 is a sample list of generic mechanical equipment that are on the safety-related equipment list. For conservatism, the direct dose on the containment pieces of generic mechanical equipment is assumed to be the fluid contact dose. Figure J.5-1 is an illustration of the point where the direct dose is calculated on a piping segment.

The secondary containment source term assumptions developed in Section J.5.3 are used for the calculation of radioactive source terms for different systems, and the fluid contact dose was calculated using QAD-P5A by following the guidelines set forth in Attachment J.C. Figures J.5-2 through J.5-3 are 6-month integrated fluid contact doses versus pipe diameter.

These curves are intended to give conservative, upper-bound direct gamma dose estimates for the qualification of the pieces of generic mechanical equipment and components in the various systems. To use these curves to calculate the direct doses on generic mechanical equipment, the following steps should be taken.

a. Identify the system on which the equipment or component is located,

b. Identify the diameter of the contaminated pipe on which the equipment is located, and

c. The 6-month integrated dose for that piece of equipment or component can be determined by reading the appropriate curve.

J.5.4 SOURCE TERM DEVELOPMENT FOR C1E/SRM EQUIPMENT OUTSIDE THE REACTOR BUILDING

The radiation level at any given location outside the reactor building following the postulated LOCA as described in Section J.3.1 is determined from the following major source contributors:

a. Direct gamma dose from radioactive piping located inside the reactor building and attenuated through the walls of the reactor building,

b. Gamma shine dose from airborne activity inside the reactor building, and

c. Gamma ray dose from airborne activity outside the reactor building.

A detailed description of the method of analysis, including the assumptions used, as well as results of the source terms is found in Attachment J.H.

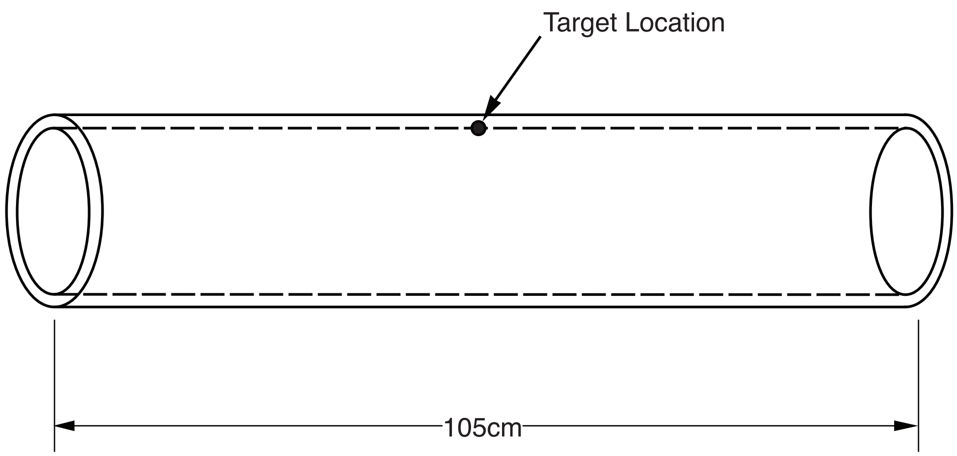
J.5.5 METHODOLOGY OF BETA DOSE ANALYSIS

The finite source volume used for the beta dose analysis in secondary containment is a sphere surrounded by a shell of sufficient thickness to stop all outside beta particles from entering the source volume. This finite spherical source volume is conservative for any generalized source shape (the dose at the center of the sphere is higher than the dose at any point of any generalized source shape of equal total volume). A discussion of this beta analysis methodology is presented in Attachment J.D.

Table J.5-1

Generic Mechanical Equipment

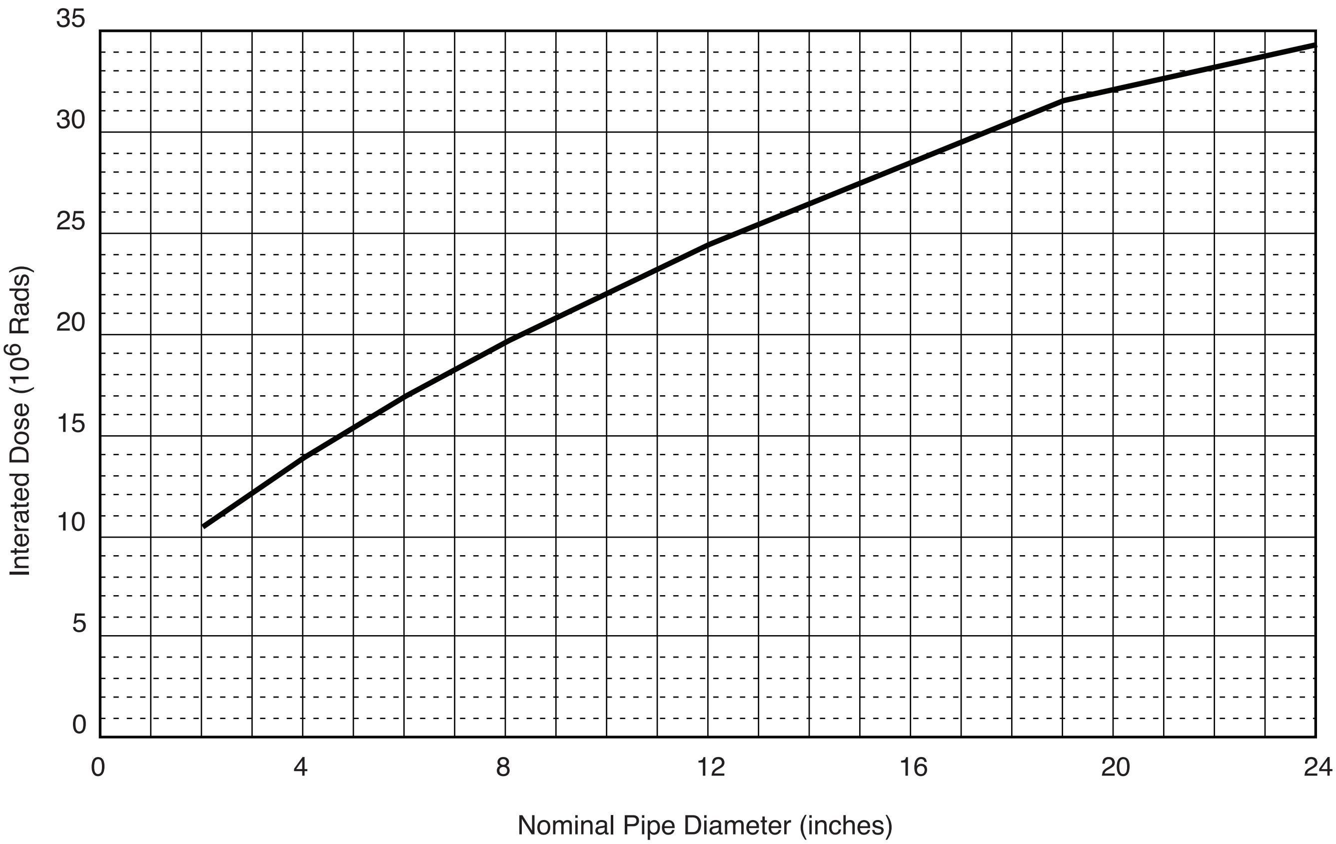
Valve packing
Lubricants
Seals
Expansion joints
Pressure relief valve
Flow element
Rupture disk
Gasket material
Conductivity element
Valve
Strainers
Steam traps
Filters (piping)
Temperature elements
Tanks
Moisture separators
Evaporator
Heat exchanger
Air washer (scrubber)
Pumps



Columbia Generating Station
Final Safety Analysis Report

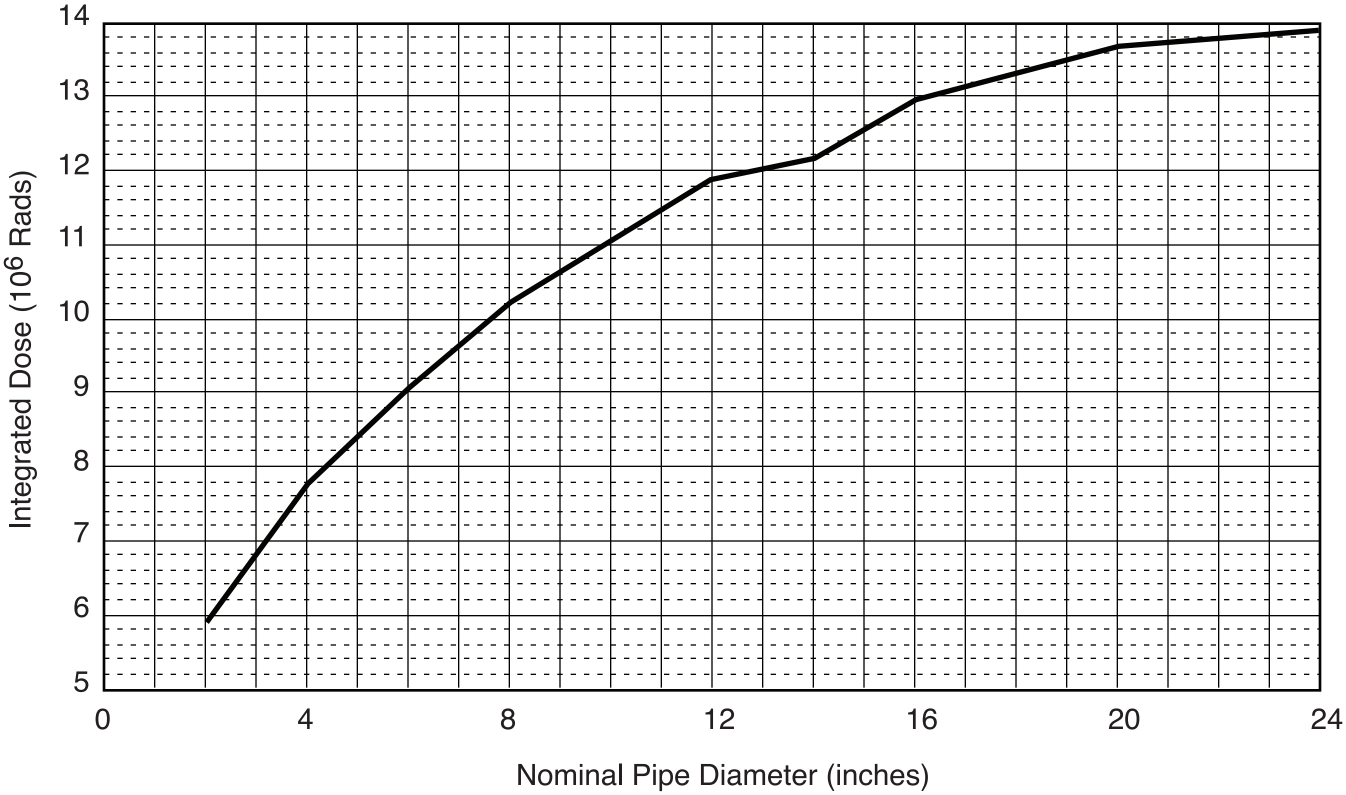
Dose Model Liquid Source

Draw. No. 970187.23 | Rev. | Figure J.5-1



Sixth Month Integrated Fluid Contact Dose for MS, RCIC (Steam) System, and MSLC System Upstream of the Header

Draw. No. 960222.63 | Rev. | Figure J.5-2



Sixth Month Integrated Fluid Contact Dose for Pipes Containing Liquid Source Term (RHR, HPCS, LPCS, RCIC Liquid Systems)

Draw. No. 960222.61 | Rev. | Figure J.5-3

J.6 RESULTS

All loss-of-coolant accident (LOCA) scenarios and accident conditions that could result in a limiting radiation environment for all the Columbia Generating Station (CGS) safety-related equipment on the C1E* list were reviewed and analyzed accordingly. Shielding (shield doors) was constructed for zones 522D, 572N, 572D, and 572H due to the radiation exposure of safety-related equipment in these zones.

In addition a shield wall was designed and installed on the southeast portion of the 501 ft el. against the bioshield wall to protect C1E* equipment from RRC piping radiation sources (normal operation) which stream through penetrations X-100A, X-105A, and X-100B.

The completeness of the safety-related equipment list has been verified. The safety-related equipment list contains all equipment required to "mitigate the consequences of an accident, bring the plant to safe shutdown conditions and provide long-term cooling capability."

Systems that could potentially contain radioactive material during and following the accident have been identified as listed in Sections J.3.2.1 and J.3.2.2.

The accident radiation doses indicated in Section J.6.1 and Table J.6-1 generated as a result of this analysis, are intended solely for the purpose of the qualification of safety-related equipment.

J.6.1 PRIMARY CONTAINMENT RADIATION RESULTS

Due to the large number of safety-related components it was deemed impractical to calculate the integrated dose to each piece of equipment. Therefore, the worst point dose from each of the major sources in the drywell and wetwell was calculated, and then summed for a conservative estimate of the total integrated dose. The dose sum of the worst-case source contributors in the drywell is 7.6×10^7 rads, but 7.4×10^7 rads is used as the worst-case dose for the equipment qualification program. All of the worst-case contributors cannot be present for a particular accident. Thus, the largest worst-case dose is calculated for the depressurized reactor coolant system. The worst-case dose is applied to safety-related equipment with an elevation within 5 ft of core midplane. Safety-related equipment in the drywell outside this elevation span is assigned a dose of 7.0×10^7 rads. In the wetwell, the maximum gamma dose above the suppression pool is 9.5×10^7 rads (see Section J.F.3 for discussion on photon energy and anticipated dose reduction of the above results). These results include the contributions from all major gamma sources within primary containment during normal operation as well as the 6-month period contribution following a postulated LOCA. Tables J.F-1 and J.F-2 give a breakdown of the integrated dose contribution from each of the

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* Environmental qualification (EQ) of safety-related mechanical (SRM) equipment has been eliminated from the overall CGS EQ program.

major gamma sources to the drywell and the wetwell. The 40-year integrated gamma doses due to normal operation are taken from Reference J.7-20.

This methodology for determining a worst-case dose for equipment in the drywell is not valid for the region inside the sacrificial shield wall or under the reactor pressure vessel. A point-specific radiation dose calculation is required for all components present in either of these two regions.

Specific calculations have been performed for equipment that was evaluated individually for total integrated dose. Results of these calculations are summarized in Reference J.7-26.

In accordance with Section 1.4(8) of Reference J.7-2, only the gamma dose need be considered for "shielded components." Since beta radiation is so readily attenuated, virtually any enclosure of sensitive components will be sufficient to classify the component as "shielded." A review of all safety-related equipment located inside primary containment determined that most C1E* equipment is sufficiently shielded against beta radiation. Thus, the beta dose contribution is excluded from the total integrated radiation doses compiled for equipment qualification purposes unless a beta-sensitive component is not adequately protected from the airborne beta environment. When required to include beta dose contributions, a finite source volume is used. The source volume is a sphere surrounded by a shell of sufficient thickness to stop all beta particles from entering the source volume. This finite spherical source volume is conservative for any generalized source volume shape (the dose at the center of the sphere is higher than the dose at any point of any generalized source shape of equal total volume). A discussion of the results is presented in Attachment J.G.

J.6.2 SECONDARY CONTAINMENT RADIATION RESULTS

The integrated direct gamma dose (40 years and 6 months LOCA - direct gamma, gamma shine, and airborne gamma) was evaluated for the worst target of all C1E* equipment in each zone and is used for qualification of all the other C1E* equipment in that zone. The 40-year integrated gamma doses (Figures J.6-1 through J.6-10) are taken from References J.7-20 and J.7-21. The direct gamma dose contribution outside primary containment due to sources inside the primary containment was investigated. Safety-related equipment located in the direct shine path through the penetrations was evaluated in Reference J.7-23. All post-LOCA radiation dose contributions to safety-related equipment from streaming through the bioshield wall penetrations were included in the radiation doses. Evaluation of bioshield wall penetrations identified radiation dose problems associated with some of those penetrations (Reference J.7-24). The post-LOCA evaluation of safety-related equipment assumed the C1E* equipment was shielded for 40-year normal operations. To adequately protect C1E* equipment a concrete wall was designed and installed for penetrations X-100A, X-105A, and X-100B. The

* Environmental qualification (EQ) of safety-related mechanical (SRM) equipment has been eliminated from the overall CGS EQ program.

remaining penetrations evaluated (Reference J.7-25) were surveyed during plant startup to confirm radiation analysis calculations.

Airborne beta doses outside containment were evaluated in accordance with the methodology described in Section J.5.5 and Attachment J.D. The beta dose contribution is excluded from the total integrated radiation doses compiled for equipment qualification purposes unless a beta sensitive component is not adequately protected from the airborne beta environment.

J.6.3 RADIATION RESULTS IN THE VITAL AREAS AND ACCESS ROUTES

Figures J.6-8 through J.6-16 present the vital areas and access routes located outside the reactor building. Figures J.6-17 and J.6-18 present the vital areas and access routes located inside the reactor building. The doses indicated on each figure are also the 6-month LOCA integrated gamma doses to be used for C1E* (safety-related) equipment qualification purposes. Table J.6-1 also presents a summary of the 6-month LOCA integrated gamma doses on all C1E* equipment located in vital areas.

Figures J.6-17 and J.6-18 show the access route in the reactor building for operation of SW-V-75AA and SW-V-75BB, the manual isolation valves for the service water to fuel pool cooling makeup water supply.

Radiation levels of vital areas and access routes were determined at selected locations outside the reactor building due to radioactive sources inside the reactor building and release of radiation activity from the reactor building elevated vent. The vital areas and access routes analyzed are consistent with those discussed in NUREG-0737, Item II.B.2 (Reference J.7-5). The radiation levels determined for the vital areas and access routes identified in Figures J.6-8 through J.6-18 are summarized in Table J.6-1. All of the vital areas and access routes have radiation levels less than the guidelines presented in NUREG-0737.

The total dose received at a vital area during a post-LOCA scenario is obtained by summing the exposure dose enroute to the vital area and the radiation dose at the vital area. These doses are listed in Table J.6-2.

The analysis completed for vital areas and access routes assumed that except for the reactor building railroad bay and on the west side of the 522-ft el. there would be no access to equipment or areas located within the reactor building during the post-LOCA scenario. The exceptions are shown in Figures J.6-17 and J.6-18 and Table J.6-2. Access to the reactor building railroad bay for 3 hr is allowed to provide the ability to fill or exchange N_2 bottles. The entry to the west side of the 522-ft el. is to allow SW-V-75AA and/or SW-V-75BB to be opened (see Section 9.1.3.2.3). These valves are readily accessible and the entire opening

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* Environmental qualification (EQ) of safety-related mechanical (SRM) equipment has been eliminated from the overall EQ program.

evolution for one of these valves would take 2.17 minutes and could be performed at 9.7 hr with the resulting exposure of 3.8 rem. Under worst-case conditions, at least one of these valves would need to be opened by 10 hr. Once a manual valve is opened, the spent fuel pool level can be controlled with the motor-operated valve from the main control room.

Table J.6-1

Six-Month Total Integrated Dose (Loss-of-Coolant Accident) to Areas Containing C1E Equipment Outside the Reactor Building

Vital Area Description	Radiation Level[a] Direct Gamma Shine +Airborne Gamma (rads)
Control room (el. 501 ft)	0.21
Technical support center	0.21
Sale area (el. 487 ft)	6.5
Nitrogen supply to ADS accumulators (el. 437 ft)	3.9
Standby service water pump valves	1.7
Remote shutdown room (el. 467 ft)	3.9
Switchgear room 1 (el. 467 ft)	3.9
Switchgear room 2 (el. 467 ft)	3.9
Radwaste control room (el. 467 ft)	3.9
Battery racks, dc battery chargers, two motor control centers (MCCs) (el. 467 ft)	3.9
Three MCCs and three switchgears (el. 437 ft)	3.9
Direct current battery charger and rack (el. 437 ft)	3.9
Diesel oil tanks (el. 437 ft)	3.9
Solid radwaste control panel and decontamination station control panel (el. 437 ft)	3.9

[a] Volume correction factors for a semi-infinite cloud were applied to the control room and technical support center. If the volume correction factors were to be applied to all areas, the integrated dose would be reduced by a minimum of fivefold.

Table J.6-2

Vital Areas and Access Route List of Radiation Exposure to Personnel During the Required Post-Loss-of-Coolant Accident Operations

Vital Area Description	Radiation Exposure: Gamma Whole Body (rem)	Thyroids (rem)[a)]	Beta Skin (rads)
Control room (el. 501 ft)[b]	0.21	0.21[c]	0.95
Technical support center[b]	0.21	0.21[c]	0.95
Security center[b]	3.1	13.4[d]	4.8
Auxiliary security center[b]	1.7		2.7
Sample analysis area (EOC)[b]	0.0013	-	-
Standby service water pump valves (cooling ponds)[e]	0.3	0.94[d]	0.46
All infrequently occupied vital areas inside the radwaste and diesel generator buildings[b]	0.13[f]	1.6[d]	0.48
Sampler for elevated release duct (roof turbine building)[e]	2.5	8.0[d]	3.8
Reactor building railroad bay (N_2 bottles)[g]	0.4	-	-
Reactor building 522-ft el. (SW-V-75AA and/or SW-V-75BB[h]	$\boxed{3.8}$	-	-
Postaccident sample area (el. 487 ft)[e]	0.36	3.2[d]	0.96
Access Routes			
All access routes inside the radwaste and diesel generator buildings[e]	0.13[f]	1.6[d]	0.48
All access routes[i] outside the radwaste and diesel generator buildings[e]	0.53	1.6[d]	0.8

Table J.6-2

Vital Areas and Access Route List of Radiation Exposure to Personnel During the Required Post-Loss-of-Coolant Accident Operations (Continued)

[a] If self-contained respiratory equipment (SCBA) is used, the thyroid dose will essentially equal the whole-body dose.

[b] Area of continuous occupancy.

[c] Assumes self-contained respiratory equipment was used by personnel during 0-3 hr post-LOCA situation.

[d] No respiratory equipment was assumed.

[e] Area occupied 0.5 hr at times after 1 hr into the LOCA.

[f] A volume correction factor for the semi-infinite cloud was included in the calculation.

[g] Assumes entry after 12 days post-LOCA for 3-hr occupancy with respiratory equipment for railroad bay portion of reactor building only.

[h] Assumes entry after 9.0 hr post-LOCA for a 2.17-minute evolution to open SW-V-75AA or SW-V-75BB with respiratory equipment and in full PC gear following access routes shown in Figures J.6-17 and J.6-18. The 2.17 minutes consists of a 1.83-minute transit (1.5 minutes in 522K and 0.33 minutes in 522H) and a 0.33-minute occupancy time in 522H.

[i] Extremely conservative analysis since the plume of airborne radioactivity cannot simultaneously cover all access routes.

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Attachment J.A

UNISOLATED LEAKING BUILDING PATH REPORT

A basic assumption to the plant shielding analysis is that the reactor isolates such that there is no radiation leakage path to the outside. A leakage path investigation was done verifying the above assumption. While performing this investigation, the total number of lines (69) penetrating the RB boundary, the associated system components and interface systems were reviewed.

The assumption eliminating the consideration of leakage is consistent with NUREG-0737, Clarification 2. This investigation assumed that containment isolation occurred prior to the egress of highly radioactive fluids. Additionally, it assumed that all safety-related equipment was available, and that all safety systems were pressurized. Therefore, at any interface, such as a heat exchanger, no potential leakage was considered if the nonradioactive system was at a higher pressure than the radioactive system. This investigation has not considered leakage from equipment seals, closed valves, or pipe rupture, except in the evaluation of the equipment and floor drain systems. The systems considered are tabulated by drawing number in Table J.A-1.

Table J.A-1

System Flow Diagrams Employed
to Perform The Review

Drawing Number	Revision	Drawing Number	Revision
M501	10	M536	12
M502	17	M537	25
M503	5	M538	9
M504	25	M539	28
M505	14	M540	15
M506	23	M541	13
M507	27	M542	4
M508	25	M543	17
M509	10	M544	10
M510	30	M545	15
M511	15	M546	10
M512	8	M547	9
M513	33	M548	14
M514	13B	M549	14A
M515	17C	M550	9
M516	20	M551	8
M517	25	M552	12
M518	14	M553	10
M519	18	M554	11
M520	15	M555	7
M521	20	M556	10
M522	6	M557	4
M523	29	M607 Sheet 1	7
M524	19	M607 Sheet 2	5
M525	19	M607 Sheet 3	3
M526	25		
M527	18		
M528	15		
M529	21		
M530	18		
M531	24		
M532	20		
M533 Sheet 1	1		
M533 Sheet 2	1		
M533 Sheet 3	1		
M534	16		
M535 Sheet 1	26		
M535 Sheet 2	21		

Attachment J.B

SOURCE TERM DEVELOPMENT AND PARAMETRIC STUDIES FOR SECONDARY CONTAINMENT

The major tools used in the development of source terms and parametric studies inside secondary containment were the ORIGEN and QAD-P5A computer codes. Descriptions of the codes are in References J.7-11 and J.7-10. ORIGEN was used to compute the activities and energies of fission products released from the reactor core. The output of ORIGEN [the time-dependent energies and activity of radioactive fission products following loss-of-coolant accident (LOCA)] was used as input to QAD-P5A to calculate the airborne, shine, and direct doses for standard geometrics as well as the basis of direct dose parametric studies.

J.B.1 RADIOACTIVE SOURCE TERMS IN SECONDARY CONTAINMENT

The ORIGEN computer code (Reference J.7-11) was used to calculate the radioactive source terms inside secondary containment for liquid-containing and gas-containing systems. The fission products at the end of fuel life were assumed to be available for release immediately following the accident. The concentrations of noble gases, halogens, and other fission products released to the gaseous and liquid sources were computed. Subsequent fission product decay and daughter product generation were then calculated for 20 time periods, covering a total period of 1 year.

The assumptions used in determining the initial distribution and leakage of radioactivity in the primary containment air and liquid space are as follows:

a. 100% of the noble gases and 50% of the halogens are distributed homogeneously within the primary containment free volume immediately following the postulated accident;

b. 50% of the halogens and 1% of the remaining fission products in the core are mixed instantaneously and homogeneously with the primary containment liquid space. The primary containment liquid space is defined as the sum of the suppression pool liquid and the reactor coolant system (RCS) liquid; and

c. The fission products available for release are defined as the total inventory generated in the equilibrium core after 1000 days at reactor power of 3556 MWt.

Assumptions a and b are NRC recommended assumptions for defining radioactivity release fractions for the qualification of safety-related equipment (Reference J.7-2) and are detailed in References J.7-32 and J.7-34.

Assumption c represents the maximum burnup level in the core and the fission products at the end of fuel life prior to radioactivity release and is conservative.

Table J.B-1 shows the gamma activity concentration at selected time periods for the liquid-containing system. The results of Table J.B-1 were used as input in the dose parametric study. Due to rapid decay of the high-energy isotopes, the average gamma energy for the gas-containing system varies from 0.8 MeV at the beginning of the accident to 0.3 MeV at 1 year after the accident.

J.B.2 AIRBORNE DOSE IN SECONDARY CONTAINMENT

The time-dependent post-LOCA activity levels as calculated by the ORIGEN computer code were used as input in the calculation of the airborne beta and gamma dose rates and integrated doses inside the cubicles in the secondary containment. The assumptions used in this analysis are as follows:

a. Activity that leaks into the secondary containment is homogeneously mixed with the secondary containment atmosphere prior to its removal from the atmosphere through the standby gas treatment system (SGTS). This is consistent with the NRC-recommended assumptions used for calculation of doses inside primary containment (Reference J.7-2 and J.7-34);

b. An SGTS flow rate of 2430 scfm was assumed to be the flow rate of the effluent air. This is the designed minimum accident flow rate (Reference J.7-35) based on one reactor building airchange per day;

c. Air that leaks out of the primary containment flows directly and totally into the secondary containment. Bypass leakage is not considered. This is conservative when considering dose in the secondary containment, since it maximizes the buildup of radioactivity in the secondary containment;

d. Geometric factors are used to convert the semi-infinite cloud gamma dose to a finite gamma dose. This assumption is used in Reference J.7-28, and is based on an average gamma ray energy of 0.733 MeV. The effect of time dependence of average gamma ray energies has been proven to be negligible; and

e. Primary containment activity leakage rate is 0.5%/day. This is consistent with the assumptions established in Reference J.7-29.

A model of the primary and secondary containment atmosphere is shown in Figure J.B-1. The activity concentration of a certain isotope inside the containment is changing due to the following three mechanisms:

a. Transport of activity due to air leakage,

b. Depletion of activity due to radioactive decay and plateout of elemental halogens inside primary containment, and

c. Increases in activity levels due to daughter product generation from fission product decay.

According to References J.7-2 and J.7-34, plateout may be modeled by an expotential removal process:

$$A(t) = A(0)\exp(-\lambda_p t)$$

Where λ_p is the removal constant due to plateout.

The first step in this calculation is to model the decay and transport of the airborne radionuclides.

General airborne activity balance in containment:

$$\frac{d}{dt}(C_{li}V_l) = \underbrace{-Q_l C_{l_i}}_{\text{leakage}} \quad \underbrace{-\lambda_i C_{li} V_l}_{\text{decay}} \quad \underbrace{-\lambda_{pi} C_{li} V_l}_{\text{plateout}} + \underbrace{\Sigma_j \lambda_j C_{lj} V_l}_{\text{growth}} \qquad \text{(J.B-1)}$$

where

C_{1i} = concentration of isotope "i"
Q_l = leakage rate from primary containment
V_1 = volume of primary containment
λ_i = radioactive decay constant of isotope "i"
λ_{pi} = plateout removal constant of isotope "i"

The term $\sum_j \lambda_j C_{1i} V_1$ reflects the growth of a given nuclide as the result of decay of parent nuclides.

The original release of nuclides consists only of halogens and noble gases. Since fission products are neutron-rich, decay of fission products proceeds toward higher atomic numbers.

In this manner, halogens will decay into noble gases, and then to higher atomic-numbered elements. Since the decay chain reaches a stable isotope after only a few decays, it can be seen that upon release of these airborne nuclides, the halogens have no significant airborne parent nuclides. This term may be neglected in the case of halogens.

Case 1 - Containment Halogens

Elemental iodine undergo plateout (Reference J.7-34) so equation (J.B-1) becomes:

$$\frac{d}{dt}(C_{1i} V_1) = -Q_1 C_{1i} - \lambda_i C_{1i} V_1 - \lambda_{pi} C_{1i} V_1 \qquad \text{(J.B-2)}$$

Solving (J.B-2) with the initial condition;

at t = 0,

$C_{1i} = C_{1i}(0)$,

$$C_{1i}(t) = C_{1i}(0) \exp\left(-\left(\frac{Q_1}{V_1} + \lambda_i + \lambda_{pi}\right) t\right) \qquad \text{(J.B-3)}$$

Particulate and organic iodine are assumed unaffected by plateout (Reference J.7-34).

Equations (J.B-3) for particulate and organic iodine may then be shown to be

$$C_{1i}(t) = C_{1i}(0) \exp\left(-\left(\frac{Q_1}{V_1} + \lambda_i\right) t\right) \qquad \text{(J.B-4)}$$

One can note, at this point, that all three iodine species have factors of

$$C_{1i}(0) \exp(-\lambda_i t)$$

in the equations. This term may be defined as

$$S_i(t) = C_{1i}(0) \exp(-\lambda_i t) \quad V_1 \qquad \text{(J.B-5)}$$

$S_i(t)$ is seen to be the total activity released into the system as a result of decay. $S_i(t)$ is independent of the transport of the nuclides. The following definitions will be made.

f_e = fraction of total iodine that are elemental
f_p = fraction of total iodine that are particulate
f_o = fraction of total iodine that are organic

Equations (J.B-3) and (J.B-4) can be combined to get

$$C_{1iH}(t) = \left[f_e \exp(-\lambda_p t) + (f_o + f_p)\right] \exp\left[-Q_1 t/V_1 \frac{S_{iH}(t)}{V_1}\right] \quad \text{(J.B-6)}$$

where

$C_{1iH}(t)$ = total iodine concentration in primary containment
$S_{iH}(t)$ = total iodine activity
λ_p = plateout constant for elemental iodine

At this point, Reference J.7-2 allows only a factor of 200 reduction for elemental iodine plateout effects.

So when

$$\exp(-\lambda_p t) = \frac{1}{200}, \text{ then } \lambda_p \text{ becomes zero.} \quad \text{(J.B-7)}$$

Defining: $t_p = \dfrac{\text{Ln}(200)}{\lambda_p}$

Equation (J.B-6) may be rewritten as

$$C_{1iH}(t) = \frac{S_{iH}(t)}{V_1} \exp\left[-Q_1 t/V_1\right] f_H(t) \quad \text{(J.B-8)}$$

Where $f_H(t)$ is defined as

$$\text{(a)} \quad f_H(t) = f_e \exp(-\lambda_p t) + f_p + f_o \qquad t \le t_p \quad \text{(J.B-9)}$$

$$\text{(b)} \quad f_H(t) = (f_e / 200) + f_p + f_o \qquad t \ge t_p$$

<u>Case 2 - Containment Noble Gases</u>

Noble gases do not undergo plateout. Daughter products are also conservatively assumed to act as noble gases. Equation (J.B-1) for noble gases becomes

$$\frac{d}{dt}(C_{1i}V_1) = -Q_1C_{1i} - \lambda_i C_{1i}V_1 + \Sigma_j \lambda_j C_{1j}V_1 \qquad \text{(J.B-10)}$$

Integrating equation (J.B-10) gives

$$C_{1i}(t) = \exp\left[-Q_1\, t/V_1\right] \exp\left[-\lambda_i t\right] (B + f_i(t)) \qquad \text{(J.B-11)}$$

where

$$f_i(t) = \int \Sigma_j \lambda_j C_{ij} \exp -\left[Q_1/V_1 + \lambda i\right] dt \qquad \text{(J.B-12)}$$

and B is a constant to be determined.

All daughter products of plated-out iodine are conservatively assumed to be re-released into the containment atmosphere as if the iodine were airborne. For the first isotope in a series (no parent nuclide), $j = 0$ and $f_o(t) = 0$.

Since $C_{1j}(t)$ has the same form as $C_{1i}(t)$, equation (J.B-12) becomes

$$f_i(t) = \int \Sigma_j \lambda_j (B + f_j(t))\, e^{(\lambda_i - \lambda_j)t}\, dt \qquad \text{(J.B-13)}$$

Equation (J.B-13) shows that the only dependence on Q_1/V_1 is that carried over from the parent isotope is $f_n(t)$. Since $f_o(t)$ is independent of Q_1/V_1, $f_i(t)$ is independent of Q_1/V_1. Equation (J.B-11) can thus be rewritten as

$$C_{1i}(t) = e^{-(Q_1/V_1)t}\, S_i(t)/V_1 \qquad \text{(J.B-14)}$$

where

$$S_i(t) = \exp\left[-\lambda_i t\right] (B + f_i(t))V_1 \qquad \text{(J.B-15)}$$

It can be seen that $S_i(t)$ is the solution to equation (J.B-10) without the leakage term. $S_i(t)$ is the activity for the total inventory of nuclides released from the reactor core. $S_i(t)$ values are determined by the use of ORIGEN. $S_i(t)$ includes radioactive decay and daughter product growth.

For a general airborne activity balance in the reactor building (secondary containment):

$$\frac{d}{dt}(C_{2i}V_2) = + \underset{\text{leakage in}}{Q_iC_{1i}} - \underset{\text{leakage out}}{Q_2C_{2i}} - \underset{\text{decay}}{\lambda_iC_{2i}V_2} + \underset{\text{growth}}{\sum_j\lambda_jC_{2j}V_2} \qquad \text{(J.B-16)}$$

where

C_{2i} = concentration of isotope "i" in the reactor building
Q_2 = leakage rate from reactor building
V_2 = volume of the reactor building

Plateout inside secondary containment is conservatively neglected.

Case 3 - Iodine Inside the Reactor Building

As in Case 1, the growth term of equation (J.B-16) is negligible. Equation (J.B-16) can be integrated to give

$$C_{2i}(t) = e^{-(Q_2/V_2 + \lambda_i)t}\left[B + \frac{Q_1}{V_2}\int e^{(Q_2/V_2 + \lambda_i)t}\, C_{1i}(t)\, dt\right] \qquad \text{(J.B-17)}$$

From equation (J.B-8), $C_{1i}(t)$ is substituted into (J.B-17)

$$C_{2i}(t) = Be^{(Q_2/V_2 + \lambda_i)t} + \frac{Q_1}{V_2}\, e^{-(Q_2/V_2 + \lambda_i)t}\int \exp\ (Q_2/V_2 + \lambda_i)t$$

$$\left(\frac{S_{iH}(t)}{V_1}\, e^{-Q_1 t/V_1}\, f_H(t)\right) dt \qquad \text{(J.B-18)}$$

Substituting equation (J.B-5) into (J.B-18) results in

$$C_{2i}(t) = B\exp\left[-(Q_2/V_2 + \lambda_i)t\right] + \frac{Q_1}{V_2}\, C_{1i}(0)\ \exp$$

$$-(Q_2/V_2 + \lambda_i)t \qquad \text{(J.B-19)}$$

$$\int \exp\left[(Q_2/V_2 - Q_1/V_1)t\right]\, f_H(t)\, dt$$

$f_H(t)$ is a complex function of time (equation J.B-9). $C_{2i}(t)$ must be solved in a series of solutions to equation (J.B-19).

For simplification, the following factors are defined

$$x = \frac{Q_2}{V_2} - \frac{Q_1}{V_1}$$

$$y = x - \lambda_p$$

Equation (J.B-19) becomes

$$C_{2i}(t) = B\exp-(Q_2/V_2 + \lambda_i)t + \frac{Q_1}{V_2 V_1} S_{iH}(t)\ \exp\ -(Q_2/V_2 + \lambda_i)t \quad \text{(J.B-20)}$$
$$\int e^{xt} f_H{}^{(t)}\ dt$$

Integrating (J.B-20) for $0 \le t \le t_p$ with the initial condition; $C_{2i}(0) = 0$ gives

(For $S_{iH}(t) = S_{iH}(0)\ e^{-\lambda_i t}$):

$$C_{2i}(t) = \frac{Q_1}{V_1 V_2} S_{iH}(t) \quad (\exp\left[-Q_1 t/V_1\right] \quad (\frac{f_e}{y} \exp\left[-\lambda_p t\right] + \frac{f_p + f_o}{x}) \quad \text{(J.B-21)}$$
$$-e^{-Q_2 t/V_2} \quad (\frac{f_e}{y} + \frac{f_p + f_o}{x}))$$

Defining

$$K_1 = (\frac{-f_e}{y} + \frac{f_p + f_o}{x})\text{; (J.B-21) becomes (for } 0 \le t \le t_p\text{):} \quad \text{(J.B-22)}$$

$$C_{2i}(t) = \frac{Q_1}{V_1 V_2} S_{iH}(t) \left(\left(\frac{f_e}{y} \exp\left[-\lambda_p t\right] + \frac{f_p + f_o}{x}\right) \exp\left[-Q_1 t/V_1\right] + K_1 \exp\left[-Q_2 t/V_2\right]\right)$$

And (for t ³ t_p): (J.B-23)

$$C_{2i}(t) = B\exp\left[-(Q_2/V_2 + \lambda_i)t\right] + \frac{Q_1}{V_1 V_2} S_{iH}(t) \exp\left[-Q_2/V_2 t\right]$$
$$\int e^{xt} \left(\frac{f_e}{200} + f_p + f_o\right) dt$$

Solving (J.B-23) gives (J.B-24)

$$C_{2i}(t) = B\exp\left[-(Q_2/V_2+\lambda_i)t\right] + \frac{Q_1}{V_1V_2}S_{iH}(t)\exp\left[-Q_2/V_2t\right]$$
$$\left(\frac{f_e/200 + f_o + f_p}{x}\right)e^{xt}$$

At t = tp (from J.B-22)): (J.B-25)

$$C_{2i}(t_p) = \frac{Q_1}{V_1\ V_2}S_{iH}(t_p)\begin{bmatrix} K_1\exp\left[-(Q_2/V_2)t_p\right] + \left(\frac{f_e}{y}\ e\ -\lambda_p t_p\right. \\ \left. + \left(\frac{f_o + f_p}{x}\right)\right)\exp\left[-Q_1t_p/V_1\right]\end{bmatrix}$$

By definition of t_p [eq. (J.B-7)]: $\exp\left[-\lambda_p t_p\right] = \frac{1}{200}$

Combining (J.B-24) and (J.B-25) at t = t_p gives (J.B-26)

$$B\exp\left[-(Q_2/V_2+\lambda_i)t_p\right] = \frac{Q_1}{V_1V_2}S_{iH}(t_p)\ \exp\left[-(Q_2/V_2)t_p\right]$$
$$\left(K_1 + \frac{f_e}{200}\left(\frac{1}{y} - \frac{1}{x}\right)\ e^{xt}\ p\right)$$

and

$$B = \frac{Q_1}{V_1V_2}\ S_{iH}(0)K_2 \qquad \text{(J.B-27)}$$

where

$$K_2 = K_1 + \frac{f_e}{200}\ \left(\frac{1}{y} - \frac{1}{x}\right)\ e^{\ xt}\ \ p \qquad \text{(J.B-28)}$$

So (J.B-24) becomes (for $t_p \leq t$)

$$C_{2i}(t) = \frac{Q_1}{V_1V_2}S_{iH}(t)\ \ (K_2 e^{-Q_2t/V_2} + \left(\frac{f_e/200 + f_o + f_p}{x}\right)e^{-Q_1t/_1} \qquad \text{(J.B-29)}$$

Equations (J.B-22) and (J.B-29) may be combined to form a general solution as follows:

$$C_{2i}(t) = S_{iH}(t)\quad F_{2H}(t)/V_2 \qquad \text{(J.B-30)}$$

where (for $0 \leq t \leq t_p$)

$$F_{2H}(t) = \frac{Q_1}{V_1}\left(K_1 \exp\left[-Q_2 t/V_2\right] + \left(\frac{f_e}{y}\exp\left[-\lambda_p t\right] + \frac{f_p + f_o}{x}\right)\exp\left[-Q_1 t/V_1\right]\right)$$

for ($t \geq t_p$) (J.B-31)

$$F_{2H}(t) = \frac{Q_1}{V_1}\left(K_2 \exp\left[-Q_2 t/V_2\right] + \left(\frac{f_e/200 + f_o + f_p}{x}\right)\exp\left[-Q_1 t/V_1\right]\right)$$

Case 4: Noble Gases Inside the Reactor Building

Equation (J.B-16) for noble gases may be rewritten as

$$\frac{d}{dt}(C_{2i}) - (Q_1/V_2)\ C_{1i} - (Q_2/V_2 + \lambda_i)\ C_{2i} + \sum_j \lambda_j C_{2j} \tag{J.B-32}$$

Integrating (J.B-32) gives (J.B-33)

$$C_{2i}(t)\ \exp\left[(Q_2/V_2 + \lambda_i)t\right] = B + \int \exp\left[(Q_2/V_2 + \lambda_i)t\right]\left(\frac{Q_1}{V_2}\ C_{1i}(t) + \sum_j \lambda_j C_{2j}(t)\right) dt$$

$C_{1i}(t)$ is found from equation (J.B-14) to be

$$C_{1i}(t) = \exp\left[-(Q_1/V_1)t\right]\ S_i(t)/V_1$$

$S_i(t)$ cannot be found analytically; hence equation (J.B-33) cannot be found analytically through this method. However, in deriving equation (J.B-14), it was shown that if all parent nuclides are transported identically, then the solution of equations consisting of transport and radioactive decay can be separated. Since the halogens are not transported in the same manner as noble gases, this is not strictly true. However, the assumption of daughter growth as if the halogens were transported will be conservative, due to the nonconsideration of the physical holdup in primary to secondary leakage of daughters of halogens.

Equation (J.B-14) may be rewritten as

$$V_1 C_{1i}(t) = S_{iN}(t)\ F_{1N}(t) \tag{J.B-34}$$

where

$$F_{1N}\ (t) = \exp[-Q_1 t/V_1] \qquad \text{(J.B-35)}$$

$S_{iN}(t)$ is the noble gas total activity term, as before. $F_{1N}(t)$ is the fraction of that activity remaining in primary containment.

Equation (J.B-16) may be modified to show the fractions of activity, rather than total isotopic activity, in secondary containment to give

$$\frac{d}{dt}(F_{2N}) = \frac{Q_1}{V_1} F_{1N} - \frac{Q_2}{V_2} F_{2N} \qquad \text{(J.B-36)}$$

Integrating equation (J.B-36) with initial conditions:

at t=0, $F_{2N} = 0$; gives

$$F_{2N}(t) = \frac{Q_1}{V_1 x} (\exp[-Q_1 t/V_1] - \exp[-Q_2 t/V_2]) \qquad \text{(J.B-37)}$$

$C_{2i}(t)$ is then found from

$$C_{2i}(t) = S_{iN}(t)\ F_{2N}(t)/V_2 \qquad \text{(J.B-38)}$$

λ_p is found in Reference J.7-2 to be determined

$$\lambda_p = K_g A_1/V_1 \qquad \text{(J.B-39)}$$

K_g is conservatively assumed to be equal to 0.05 cm/sec (Reference J.7-34).

A_1 is the surface area inside the drywell = $3.2 \times 10^7 cm^2$ (Reference J.7-33).

$V_1 = 5.68 \times 10^9 cm^3$ (Reference J.7-36)

$\lambda_p = 1.01\ hr^{-1}$

To calculate the airborne gamma dose rate inside the secondary containment, the method as described in Reference J.7-28 is used:

$$D_{\gamma\infty} = \sum_{i=1}^{n} 0.25 \overline{E}_{\gamma} i\ (C_{2i}\ \text{noble gas} + C_{2i}\ \text{halogen}) \qquad \text{(J.B-40)}$$

$$D_\gamma = \frac{D\gamma\infty}{GF} \tag{J.B-41}$$

$$GF = \frac{1173}{V^{0.338}} \tag{J.B-42}$$

where

$D_{\gamma\infty}$ = semi-infinite gamma cloud dose rate (rads/sec)

$\overline{E}\gamma_i$ = average gamma energy of the isotope (MeV)

C_{2i} = activity concentration inside secondary containment (Ci/m^3)

GF = geometric factor used to scale the semi-infinite gamma cloud dose to a finite cloud dose

V = volume of the finite cloud (ft^3)

By taking $S_i(t)$ from ORIGEN output and using equations (J.B-31) and (J.B-37) to calculate $F_2(t)$, the total gamma dose in secondary containment can be computed by using equations (J.B-40) through (J.B-42).

The airborne semi-infinite cloud gamma dose rates are shown in Figure J.B-2. As can be observed from the figures, the gamma doses inside secondary containment reach their peaks at around three days after the accident, and decay slowly thereafter due to the depletion of radioactivity by radioactive decay and removal through the SGTS.

The geometric factor in equation (J.B-42) is developed in Reference J.7-28 for average gamma energies of 0.733 MeV. There has been a concern that this geometric factor may vary appreciably with time due to the faster decay rate of the high energy isotopes. The average gamma energy during various time periods following the accident were computed and the results show that the average gamma energy varies from 0.3 MeV to 0.8 MeV. As discussed in Reference J.7-31, the geometric factor changes by less than 5% within that energy range. It is therefore concluded that the change in the geometric factors with time is negligible, and that equation (J.B-42) can be used to calculate the finite cloud gamma dose inside the secondary containment.

J.B.3 PARAMETRIC STUDIES FOR DIRECT PIPING DOSE

The purpose of the parametric study was to identify the parameters which have a significant affect on the radiation dose rates. The computer code QAD-P5A was used to develop a correlation scheme for the significant parameters such that a simplified procedure for calculating radiation dose rates for complex source and receptor geometries can be developed. The dose rate at a target distance of 8 ft radially outwards from the centerline of an 8-in. schedule 40 pipe, infinitely long (standard pipe) was first calculated and defined as the standard dose rate. A parametric study was then performed to investigate the effects of the variation of parameters such as pipe length, pipe diameter, shield thickness, and target locations on the dose rate. The results of this parametric study were then correlated as a set of correction factors to the standard dose rate. A simplified procedure was developed to calculate the dose rates and cumulate doses for the multitude of source-target configurations by using these correction factors.

J.B.3.1 Functional Dependence of Various Parameters on Secondary Containment Dose Rates

The gamma ray energy flux from a line source "S_L" to a detector point "P" (see Figure J.B-3) is shown in Reference J.7-30 as

$$\phi = \frac{BS_L}{4\pi r} {}_0\int^{\theta 1} \exp - b_1 \operatorname{Sec} \theta_{d\theta} - \int^{\theta 2} \exp - b_1 \operatorname{Sec} \theta_{d\theta} \qquad \text{(J.B-43)}$$

where

ϕ = uncollided gamma ray flux (photons/cm^2 - sec)
b_1 = total attentuation through shield
S_L = source strength of line source (photons/cm sec)
B = buildup factor
θ = angle subtended by the length of the line source (see Figure J.B-3)

The source strength "S_L" is a function of the volume of liquid inside the pipe segments, which is also a function of the diameter and volume of the pipe. The angles "θ1" and "θ2" are also functions of "a/r" and "b/r," respectively (see Figure J.B-18 for definition of "a/r" and "b/r" respectively). Therefore, the functional dependence of gamma ray dose rates on the various parameters can be represented by the following equation:

$$\phi = \phi_o * F_D * F_R * F_L \left[(a/r, b_1) + F_L (b/r, b_1)\right] \qquad \text{(JB-44)}$$

where

ϕ_o = base gamma ray flux for standard pipe
F_D = pipe diameter correction factor
F_R = radial distance correction factor
F_L = $(a/r, b_1)$ = pipe length correction factor

J.B.3.2 Parametric Study Procedures

The procedure for performing this parametric study is documented as follows:

a. Calculate the dose rate at a target distance of 8 ft from the centerline of an 8-in. schedule 40 pipe infinitely long (standard pipe);

b. Perform parametric studies on the variation of dose rates with

   1. Radial distance from the pipe centerline,
   2. Length of the pipe,
   3. Nominal pipe diameter,
   4. Time, and
   5. Axial position along the pipe;

c. Correlate the results of the parametric study by a set of geometric correction factors;

d. Develop a procedure for calculating dose rates by using the correction factors; and

e. Verify the correlation scheme by calculating the dose rates at different target locations due to source piping of varied geometries through the use of QAD-P5A computer code, and compare the results to those obtained by using the procedure developed in step d.

J.B.3.3 Direct Dose Parametric Study Results Inside Secondary Containment

The standard pipe gamma dose rate and integrated dose curves for the different systems having different source term assumptions (defined in Section J.5.3.2) are shown in Figures J.B-4 through J.B-11. The various correction factors were calculated by the following correlation.

$$F_R(r) = \frac{\text{Dose rate at a radial distance "r" from an infinitely long 8-in. sch 40 pipe}}{\text{Dose rate at a radial distance of 8 ft from an infinitely long 8-in. sch 40 pipe}}$$

$$F_L(\ell) = \frac{\text{Dose rate at a radial distance of 8 ft from an 8-in. sch 40 pipe of length "}2\ell\text{"}}{\text{Dose rate at a radial distance of 8 ft from an infinitely long 8-in. sch 40 pipe}}$$

$$F_D(d) = \frac{\text{Dose rate at a radial distance of 8 ft from an infinitely long sch 40 pipe of nominal diameter "d"}}{\text{Dose rate at a radial distance of 8 ft from an infinitely long 8-in. sch 40 pipe}}$$

The above mentioned correction factors for liquid system source terms are shown in Figures J.B-12, J.B-13, and J.B-14. The correction factor curves for gaseous source terms are shown in Figures J.B-15, J.B-16, and J.B-17.

J.B.3.4 Correction Factor Method of Determining Direct Doses in Secondary Containment

Using the parametric curves from Section J.B.3.3, one obtains dose rates at varied radial distances (between 2 ft to 40 ft) from varied pipe diameters (between 2 in. to 24 in.) of varied lengths (between 2 ft to infinity) at any given time period within 1 year. The step-by-step procedure for calculating direct dose is as follows:

a. Identify a/r, b/r parameters and obtain pipe length correction factor F_L from Figure J.B-13 or J.B-16, depending on the system being considered. (See Figure J.B-18 for definition of "a/r" and "b/r");

b. Obtain the standard dose rate from the standard dose rate curve for time "t" desired;

c. Obtain the pipe diameter correction factor $F_D(d)$;

d. Obtain radial distance correction factor $F_R(r)$; and

e. The dose rate for the given pipe segment can be computed by

Dose Rate = (Standard Dose Rate) $(F_R)(F_D)(F_L)$.

Table J.B-2 compares the results for dose rate of 17 different pipe geometry and target locations as calculated using the correction factor method to those calculated by using the computer code QAD-P5A. It was observed that the biggest difference in results between the two methods is less than 10%. It is concluded that the correction factor method is adequate for calculating direct dose.

Table J.B-1

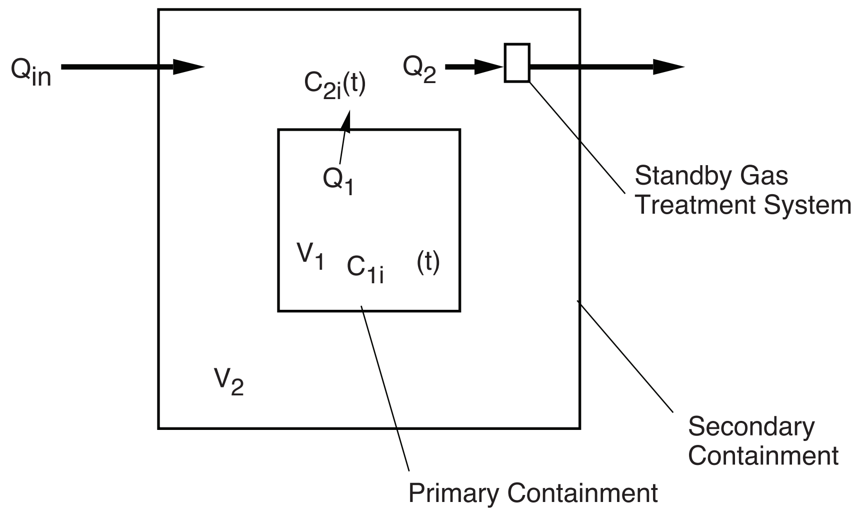
Gamma Energy Concentration (photons/sec-cm^3) in Liquid-Containing Systems

	Gamma Energy (MeV)											
Time	0.30	0.63	1.10	1.55	1.99	2.38	2.75	3.25	3.70	4.22	4.70	5.25
0 min	1.32E+09	7.25E+09	2.33E+09	1.63E+09	1.28E+08	1.00E+08	1.33E+08	3.61E+07	1.90E+07	2.82E+07	4.58E+07	3.39E+05
2 min	1.17E+09	7.17E+09	2.03E+09	6.17E+08	1.25E+08	4.54E+07	4.88E+07	2.42E+07	1.03E+07	7.17E+06	1.02E+07	2.10E+05
6 min	1.06E+09	6.92E+09	1.85E+09	5.63E+08	1.22E+08	1.99E+07	1.55E+07	1.62E+07	7.34E+06	8.50E+05	5.84E+05	8.09E+04
20 min	9.71E+08	6.21E+09	1.69E+09	5.00E+08	1.14E+08	1.30E+07	8.21E+06	9.17E+06	5.25E+06	2.03E+04	3.55E+03	2.86E+03
1 hr	8.84E+08	4.75E+09	1.41E+09	3.84E+08	1.01E+08	7.71E+06	3.36E+06	2.60E+06	2.17E+06	1.43E+00	3.66E-01	2.01E-01
3 hr	8.50E+08	2.65E+09	9.92E+08	2.28E+08	7.75E+07	2.77E+06	3.61E+05	3.74E+05	1.58E+05	3.26E-03	1.55E-03	9.71E-04
9 hr	9.04E+08	1.29E+09	5.00E+08	1.05E+08	3.88E+07	7.71E+05	9.04E+03	2.35E+04	6.17E+01	3.26E-03	1.55E-03	9.71E-04
1 day	8.09E+08	7.17E+08	1.39E+08	3.73E+07	8.84E+06	6.17E+05	1.30E+03	6.29E+01	5.17E-03	3.26E-03	1.54E-03	9.71E-04
3 days	5.54E+08	2.71E+08	1.79E+07	1.91E+07	9.34E+05	5.71E+05	1.10E+03	3.56E+01	5.17E-03	3.26E-03	1.54E-03	9.67E-04
9 days	3.16E+08	1.22E+08	4.58E+06	1.36E+07	5.71E+05	4.29E+05	1.09E+03	3.46E+01	5.13E-03	3.22E-03	1.53E-03	9.59E-04
30 days	5.42E+07	6.46E+07	1.73E+06	4.38E+06	2.67E+05	1.48E+05	1.05E+03	3.31E+01	4.96E-03	3.12E-03	1.48E-03	9.29E-04
60 days	8.17E+06	4.42E+07	9.29E+05	1.03E+06	1.49E+05	3.94E+04	9.92E+02	3.13E+01	4.71E-03	2.98E-03	1.41E-03	8.88E-04
90 days	3.99E+06	3.45E+07	7.13E+05	3.60E+05	1.14E+05	1.75E+04	9.38E+02	2.96E+01	4.54E-03	2.86E-03	1.35E-03	8.50E-04
120 days	3.25E+06	2.77E+07	6.25E+05	2.19E+05	1.00E+05	1.26E+04	8.88E+02	2.80E+01	4.38E-03	2.75E-03	1.30E-03	8.17E-04
150 days	2.90E+06	2.27E+07	5.79E+05	1.83E+05	9.17E+04	1.11E+04	8.38E+02	2.64E+01	4.21E-03	2.66E-03	1.26E-03	7.92E-04
180 days	2.64E+06	1.89E+07	5.42E+05	1.69E+05	8.50E+04	1.04E+04	7.92E+02	2.50E+01	4.08E-03	2.57E-03	1.22E-03	7.67E-04

Table J.B-2

Comparison of Direct Dose Rate Results

Target Location and Pipe Geometry					Dose Rate Results			
Pipe Diameter (cm)	Pipe Length (cm)	Target Location r (cm)	Target Location a (cm)	Target Location b (cm)	Time After Accident (hr)	Correction Factor Method (rad/hr)	Computer Results (rad/hr)	Difference (%)
6	800	548.6	570	230	24	52.7	53.3	-1.1
6	800	91.4	720	80	24	484	479	+1.0
6	800	1006.8	650	150	24	15.3	16.1	+5.0
8	800	548.6	570	230	24	77.0	80.4	-4.23
8	800	391.4	720	80	24	105.0	110.0	-4.5
8	800	1066.8	650	150	24	22.4	24.4	-8.2
2	700	100.0	600	100	720	5.36	5.32	0.75
2	700	1066.8	600	100	720	0.159	0.146	8.9
2	700	100	-900	1600	720	0.0126	0.0123	2.3
2	700	1066.8	200	900	720	0.128	0.124	3.2
12	400	1066.8	-400	800	720	1.14	1.21	-5.8
12	400	100	350	50	720	72.5	71.4	1.5
12	400	609.6	350	50	720	4.66	4.93	-5.5
12	400	1066.8	350	50	720	1.57	1.73	-9.3
10	600	304.8	-243.8	548.6	0.0333	554	539	2.7
10	600	121.9	450	150	0.0333	7617	7396	-3.0
10	600	1005.8	450	150	0.0333	258	280	-7.9



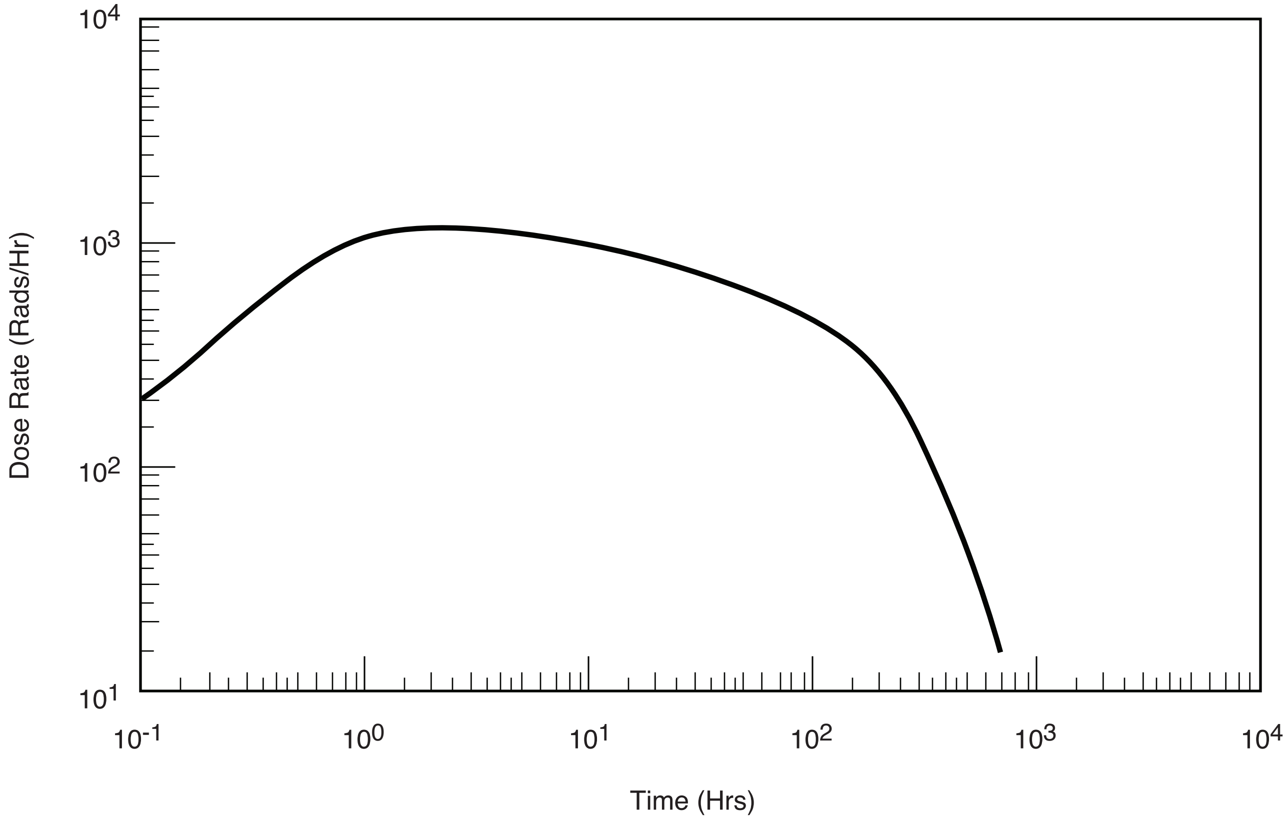
Q_2 = Air Leakage Rate from Secondary Containment (m^3/sec)
Q_1 = Air In-leakage Rate from Primary Containment (m^3/sec)
V_2 = Volume of Secondary Containment (m^3)
V_1 = Volume of Primary Containment (m^3)
i = Nuclide Index
$C_{2i}(t)$ = Activity Concentration in Secondary Containment (Ci/m^3)
$C_{1i}(t)$ = Activity Concentration in Primary Containment (Ci/m^3)
Q_{in} = Clean Air In-leakage Rate (m^3/sec)

Columbia Generating Station
Final Safety Analysis Report

Model of the Primary and Secondary Containment

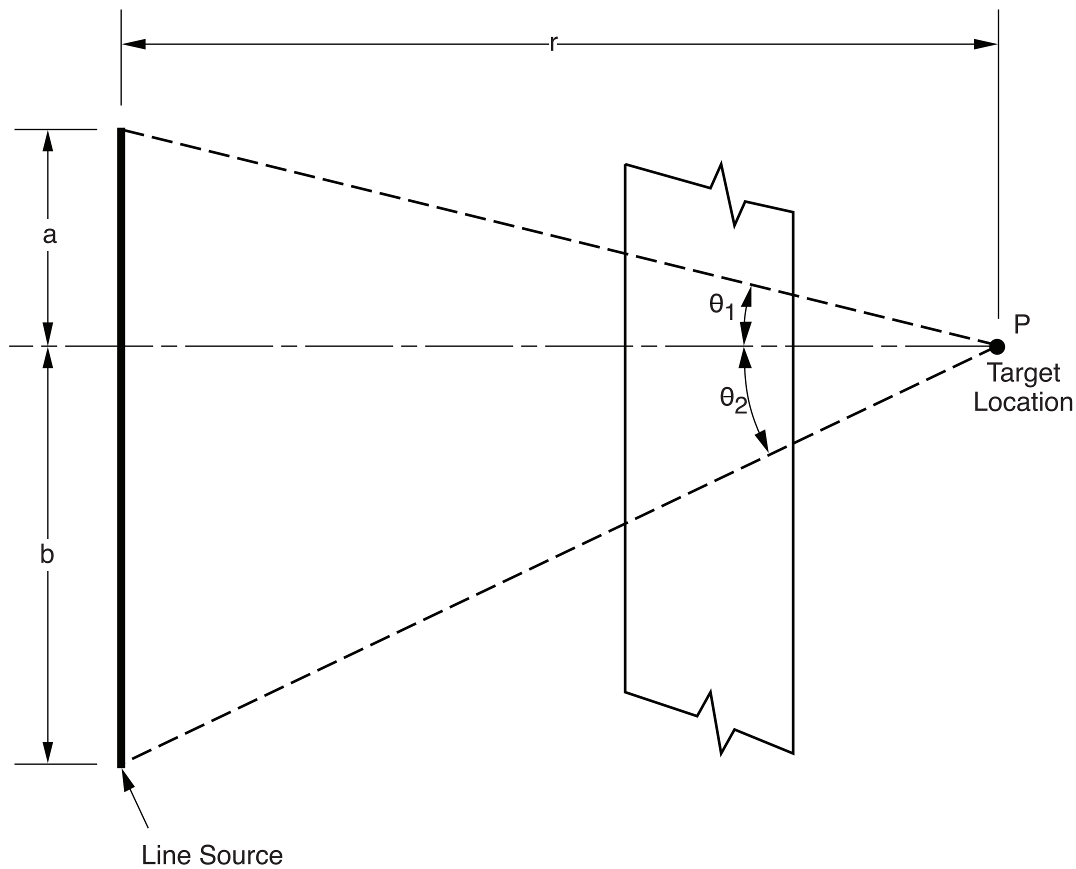
Draw. No. 970187.24	Rev.	Figure J.B-1

Form No. 960690



0.5%/Day Primary Containment Leakage Rate

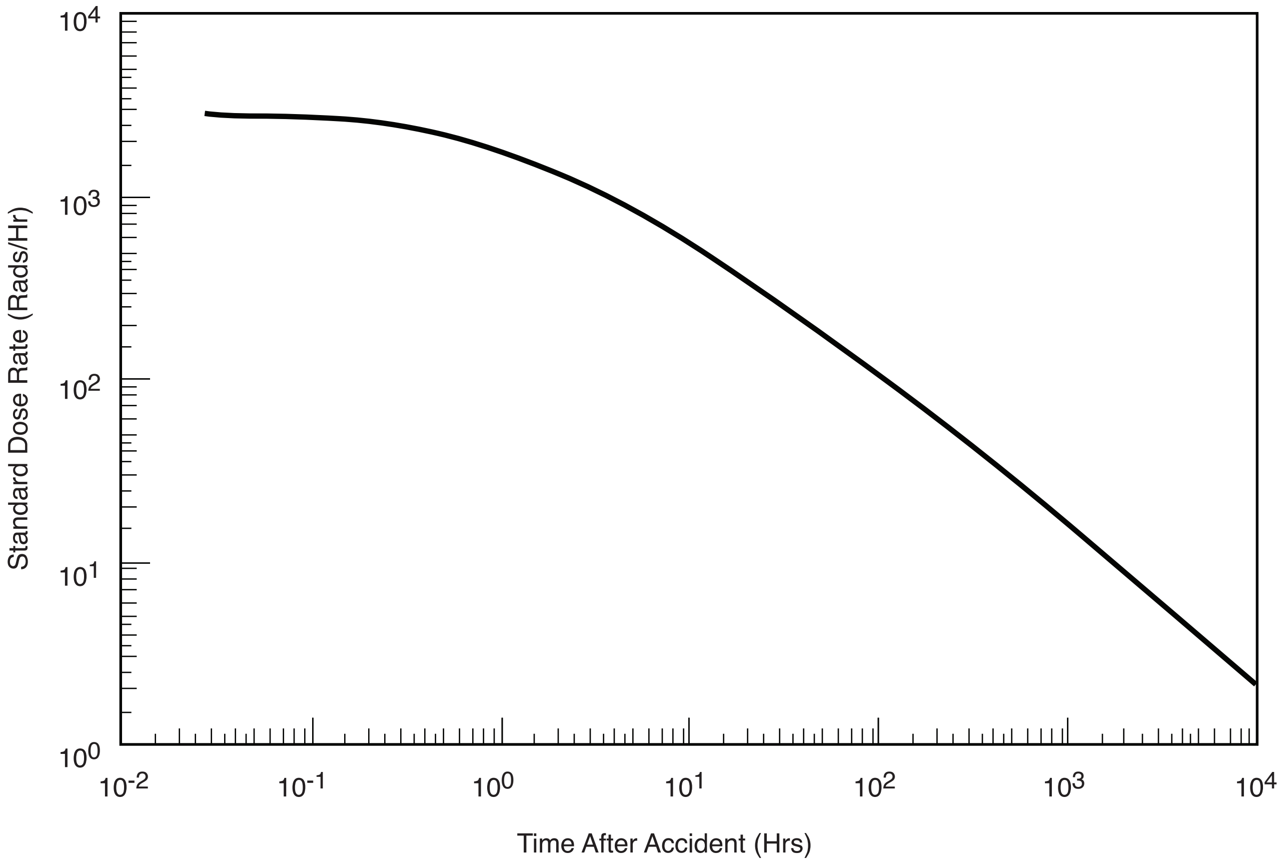
Columbia Generating Station Final Safety Analysis Report	Time-Dependent Gamma Dose Rate for a Semi-Infinite Cloud of Fission Products at Secondary Containment Concentrations	
	Draw. No. 970187.25	Rev.	Figure J.B-2



$$b_i = \sum_i \mu_i t_i$$

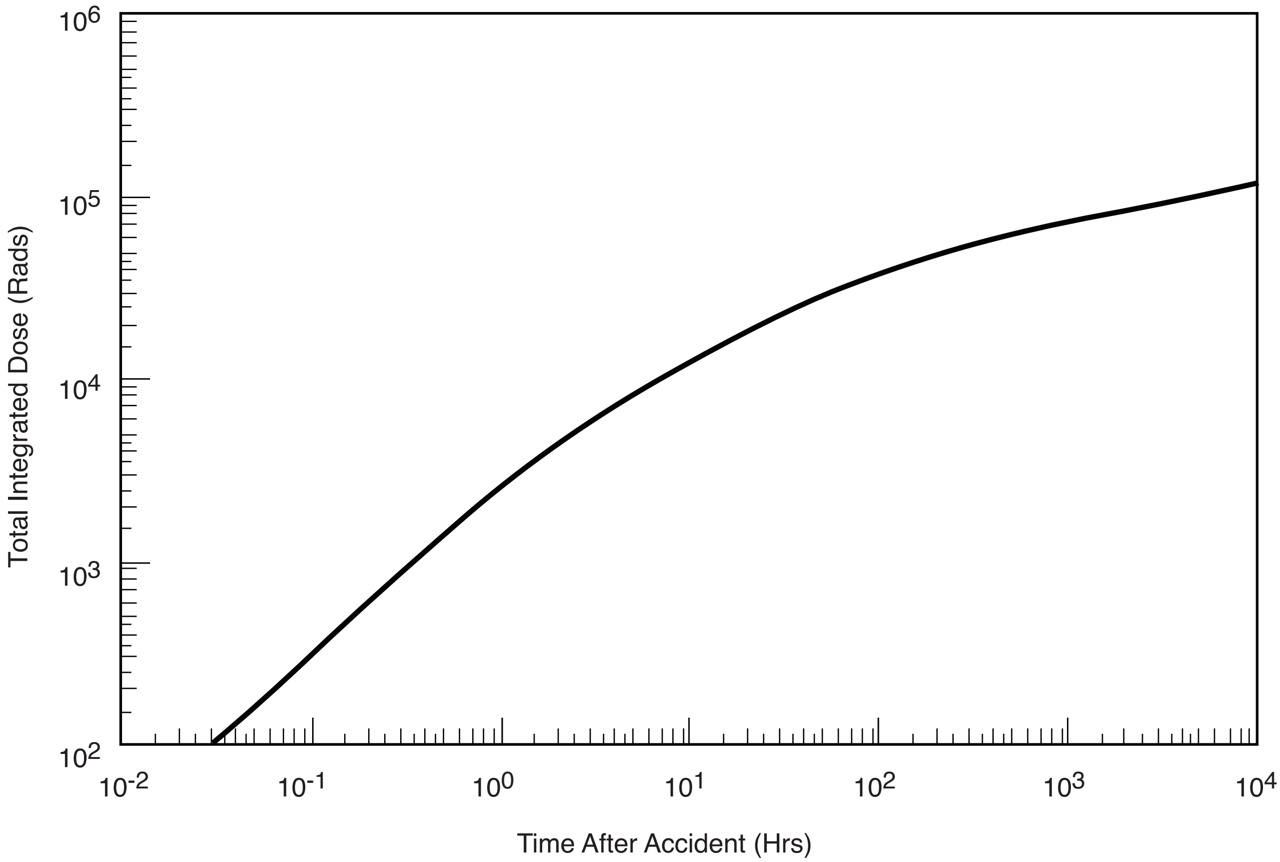
Illustration of Parameters Used in the Shielding Equation

Draw. No. 970187.26 Rev. Figure J.B-3

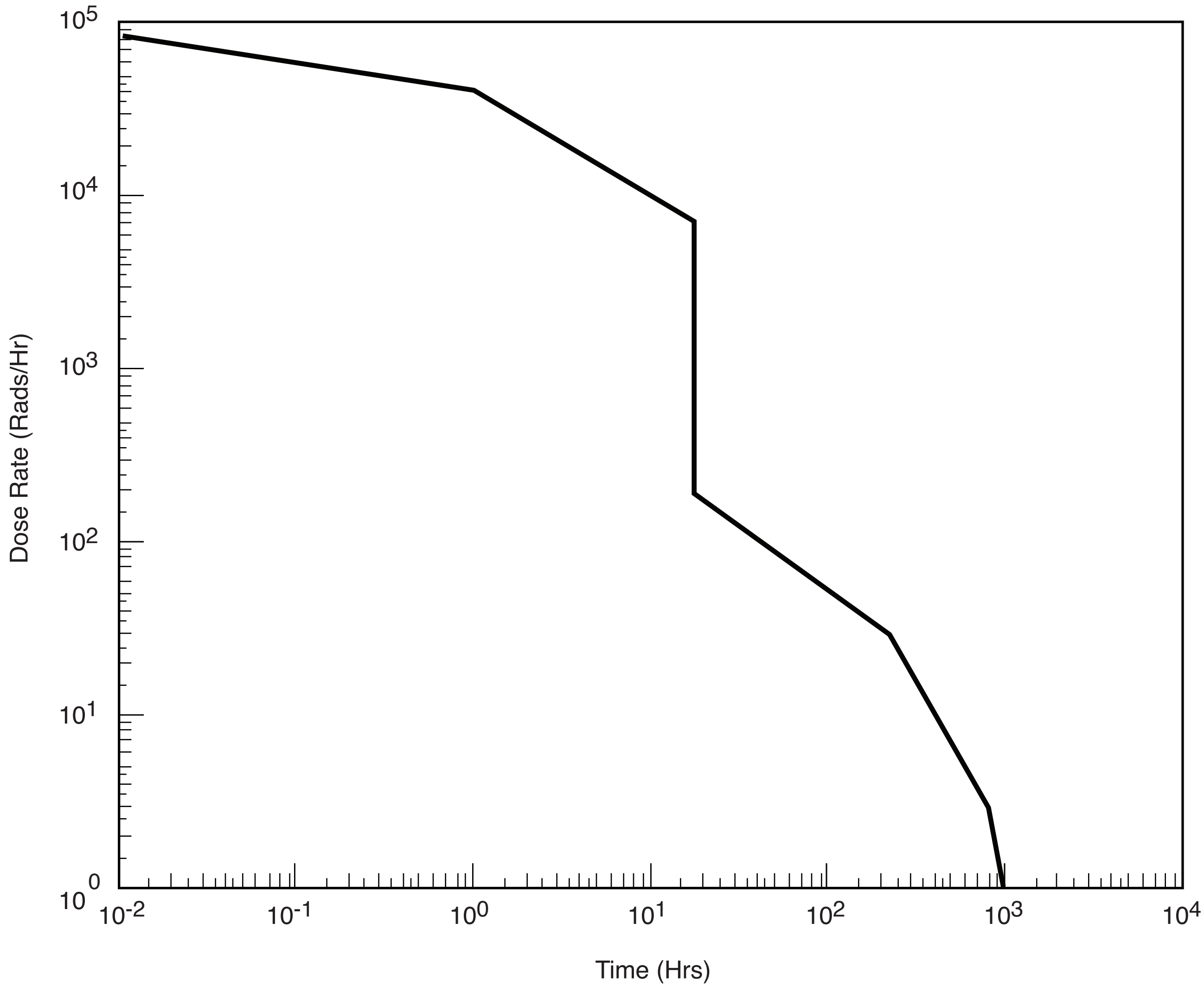


Columbia Generating Station Final Safety Analysis Report	**Standard Gamma Dose Rate Curve for Liquid Containing Systems (RCIC Liquid System and RHR System)**
	Draw. No. **970187.27** Rev. Figure **J.B-4**

Form No. 960690

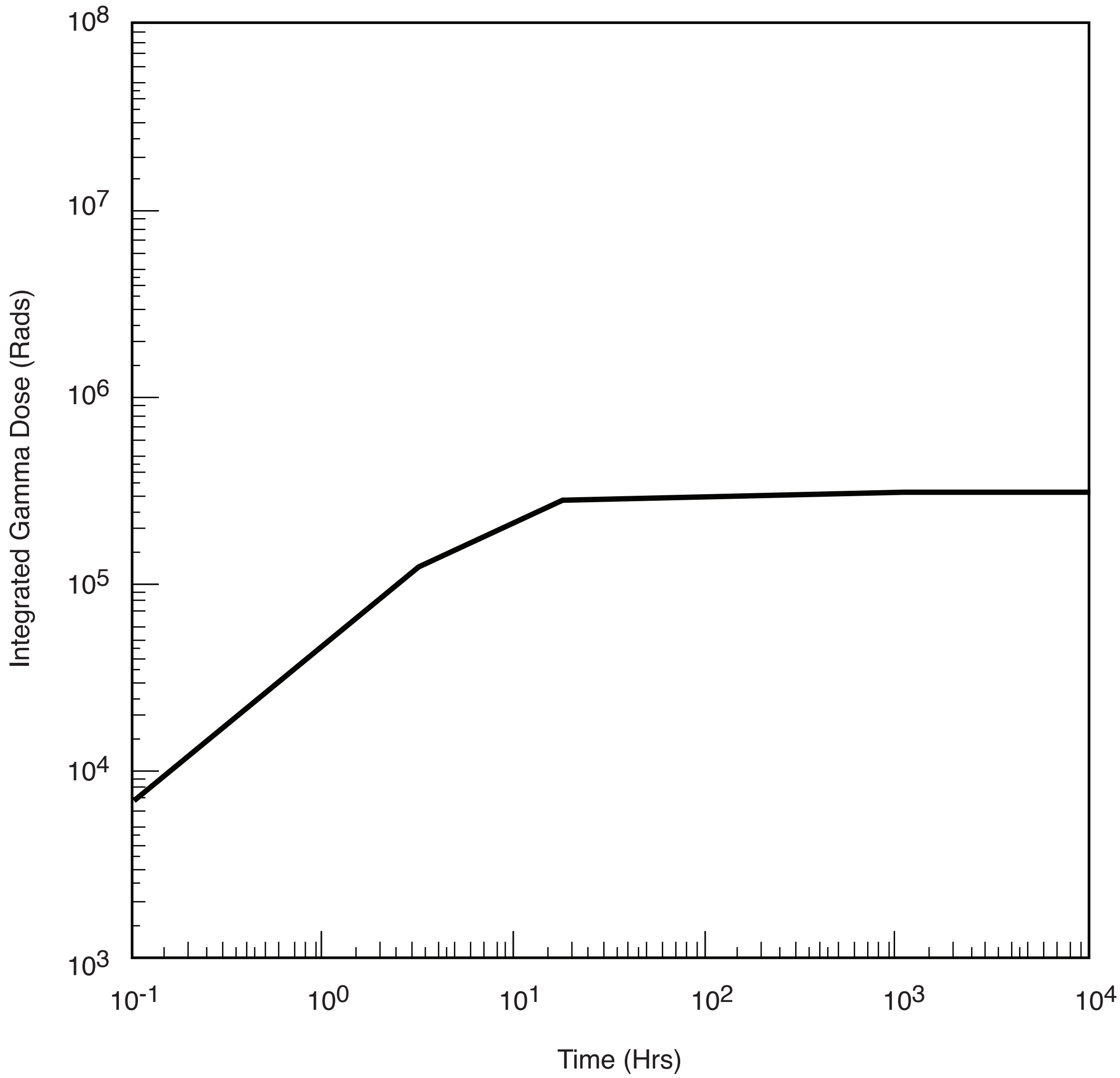


Columbia Generating Station **Final Safety Analysis Report**	**Standard Integrated Gamma Dose Rate Curve for Pipes in Liquid Containing Systems (RCIC Liquid System and RHR System)**
	Draw. No. **970187.28** Rev. Figure **J.B-5**



Standard Gamma Dose Rate Curve for Pipes in the RCIC Steam System and MSIV-LCS Steam System Before the Header

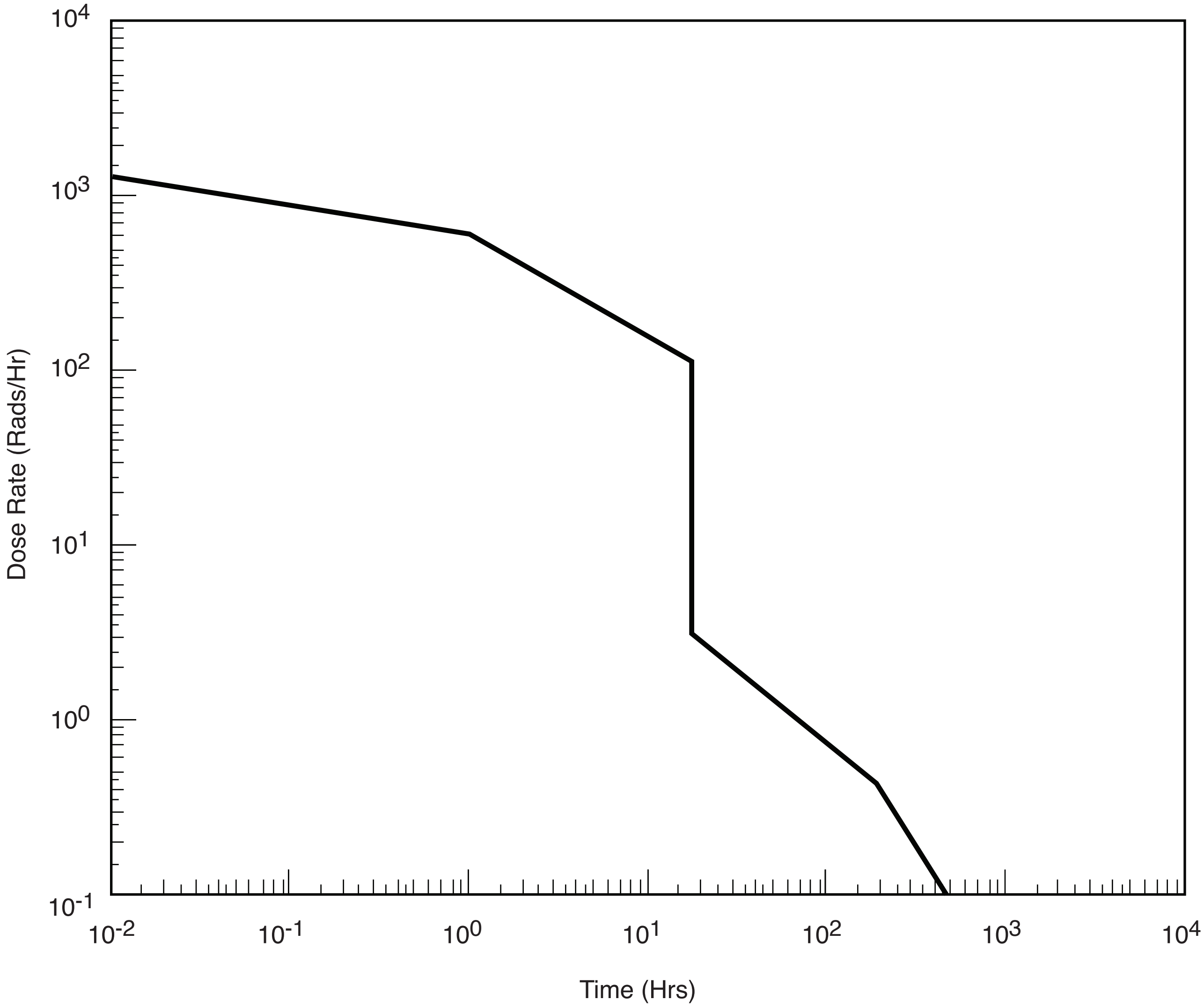
Draw. No. 970187.29 | Rev. | Figure J.B-6



Columbia Generating Station
Final Safety Analysis Report

Standard Integrated Gamma Dose Curve for Pipes in the RCIC Steam System and MSIV-LCS Steam System Before the Header

Draw. No. 970187.30	Rev.	Figure J.B-7



Columbia Generating Station
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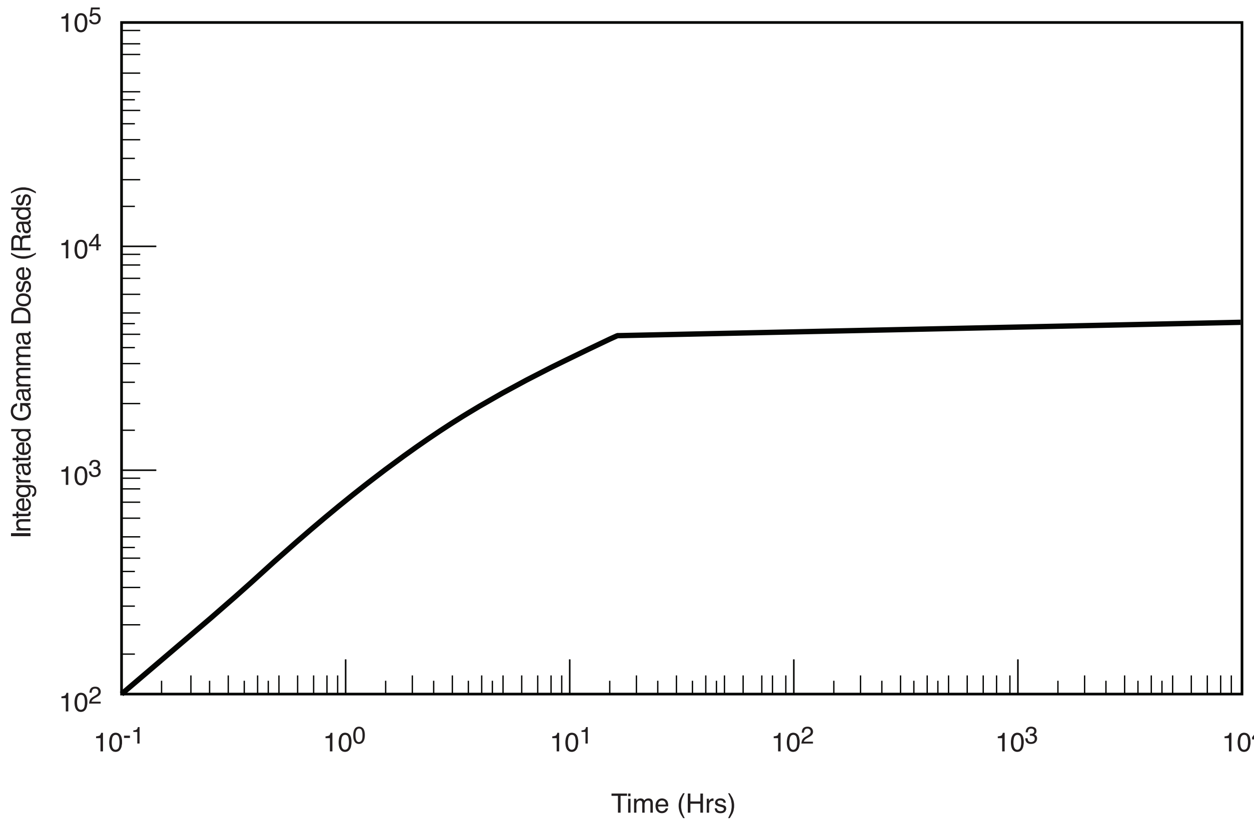
Standard Gamma Dose Rate Curve for Pipes in the MSIV-LCS Steam System After the Header

Draw. No. **970187.31** Rev. Figure **J.B-8**



Standard Integrated Gamma Dose Curve for Pipes in the MSIV-LCS Steam System After the Header

Draw. No. 970187.32 | Rev. | Figure J.B-9

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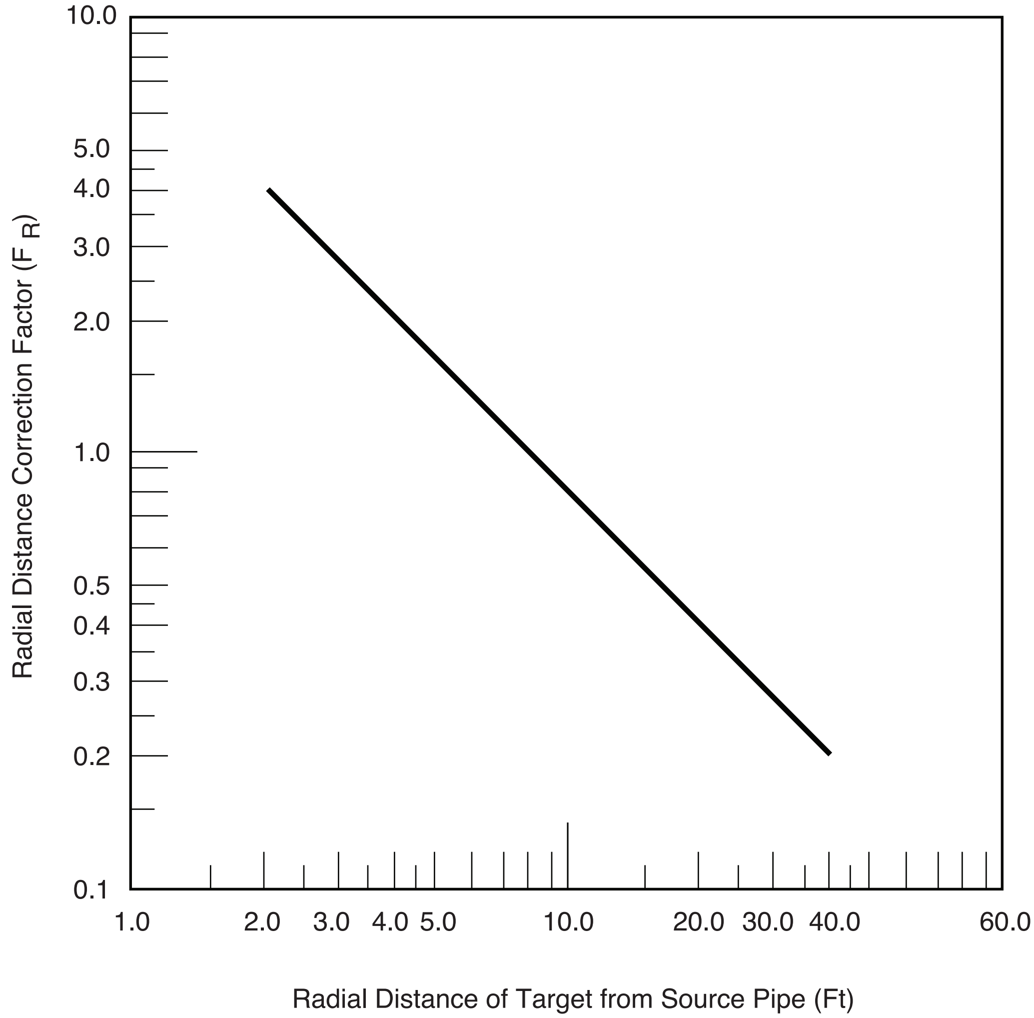
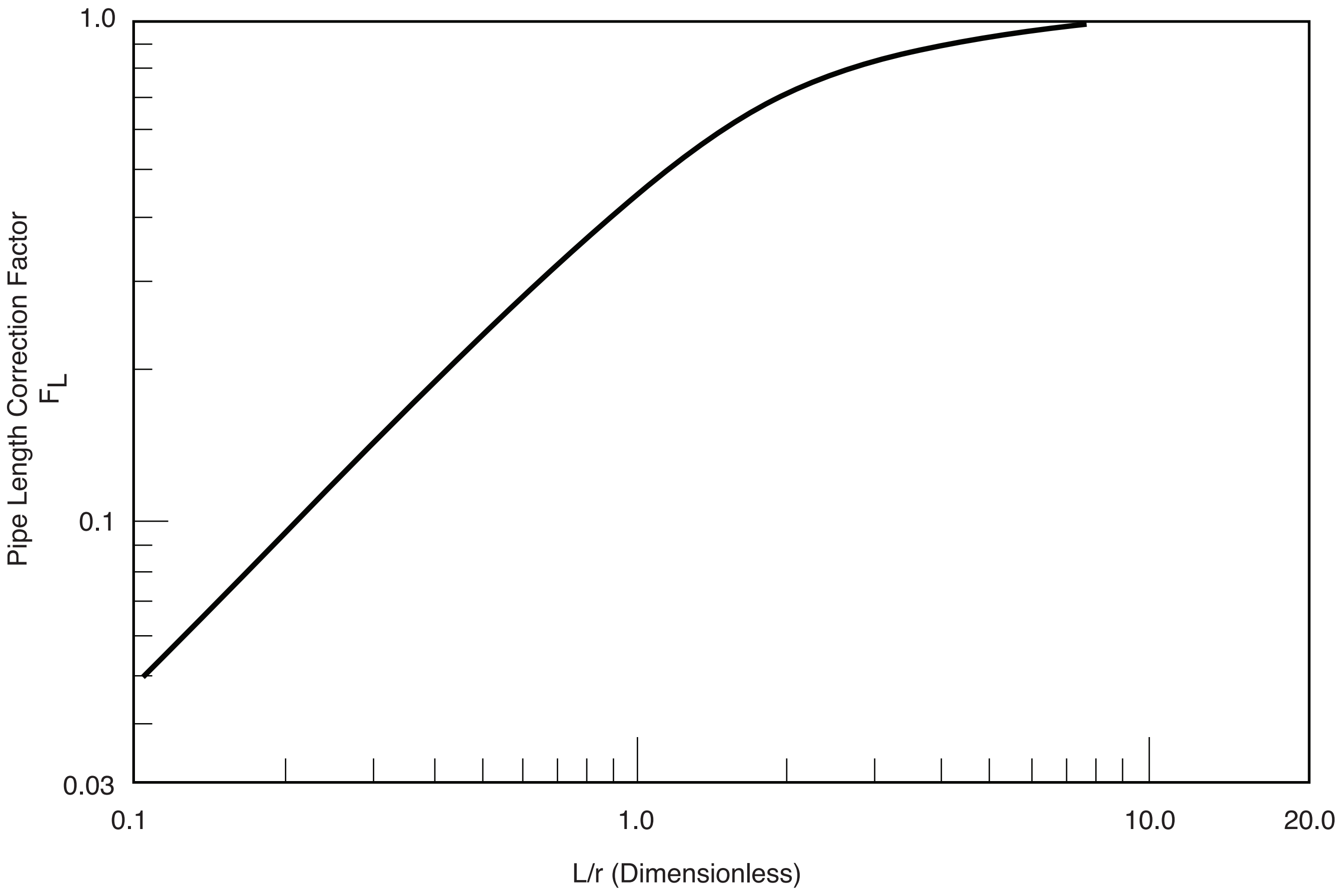


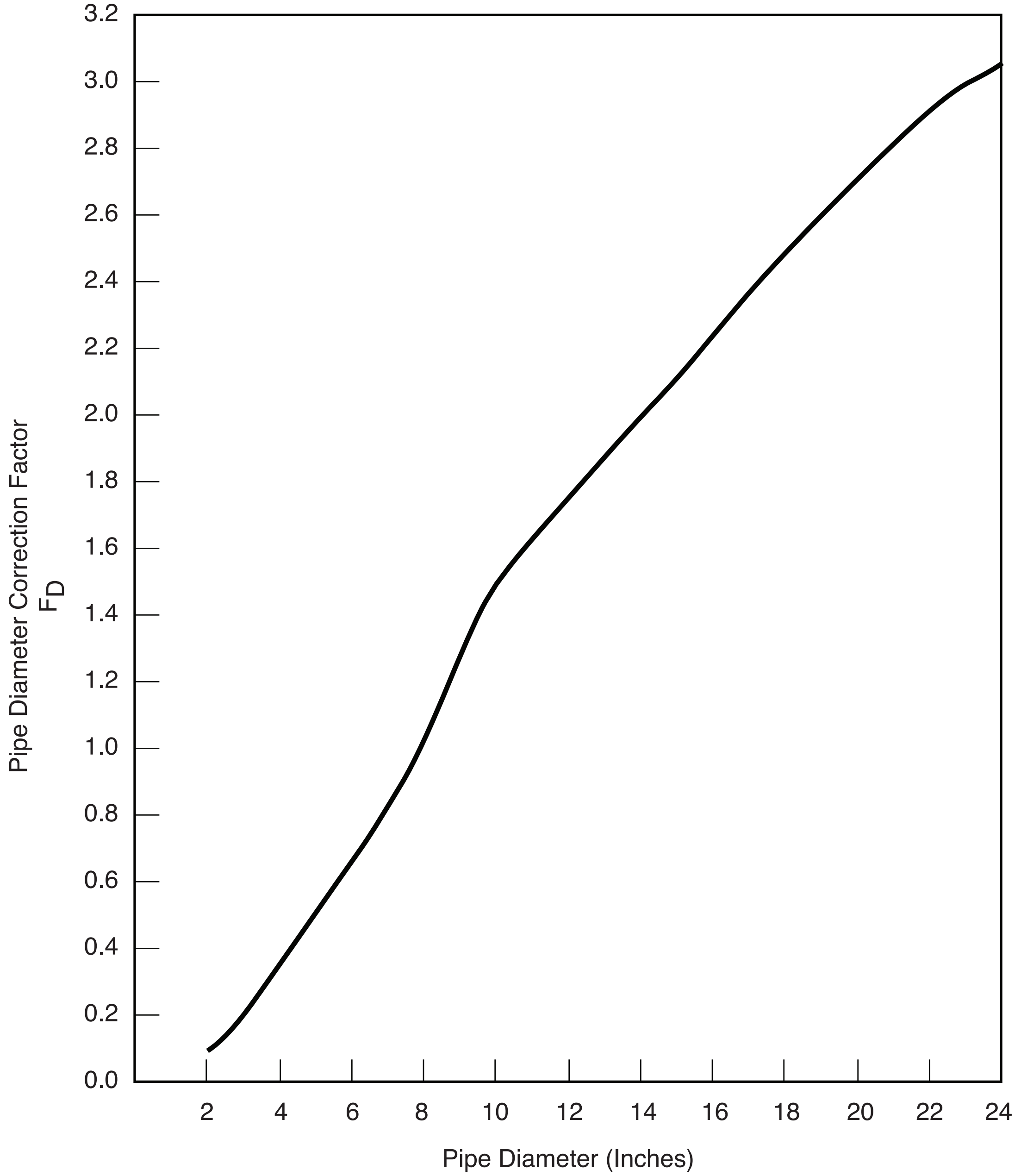
Columbia Generating Station Final Safety Analysis Report	Radial Distance Correction Factor for Liquid Sources		
	Draw. No. 970187.35	Rev.	Figure J.B-12



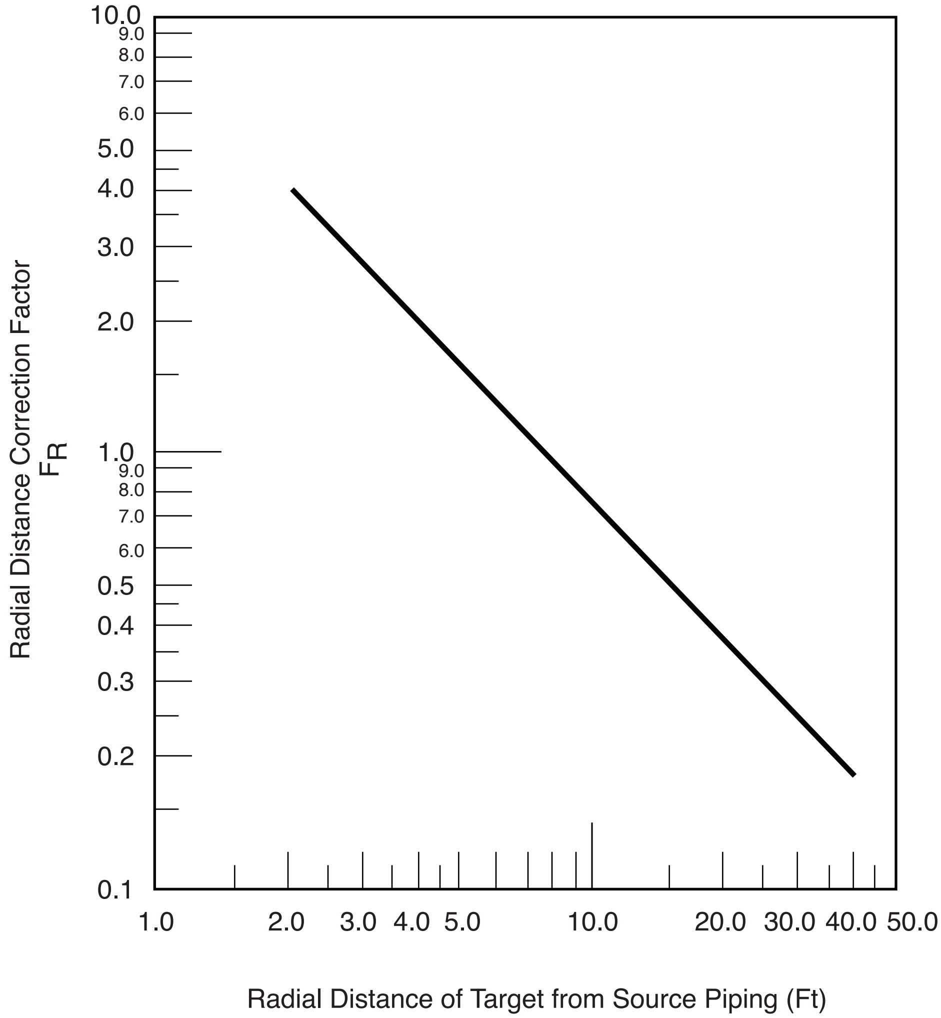
Columbia Generating Station
Final Safety Analysis Report

Pipe Length Correction Factor for Liquid Sources

Draw. No. 970187.36 | Rev. | Figure J.B-13

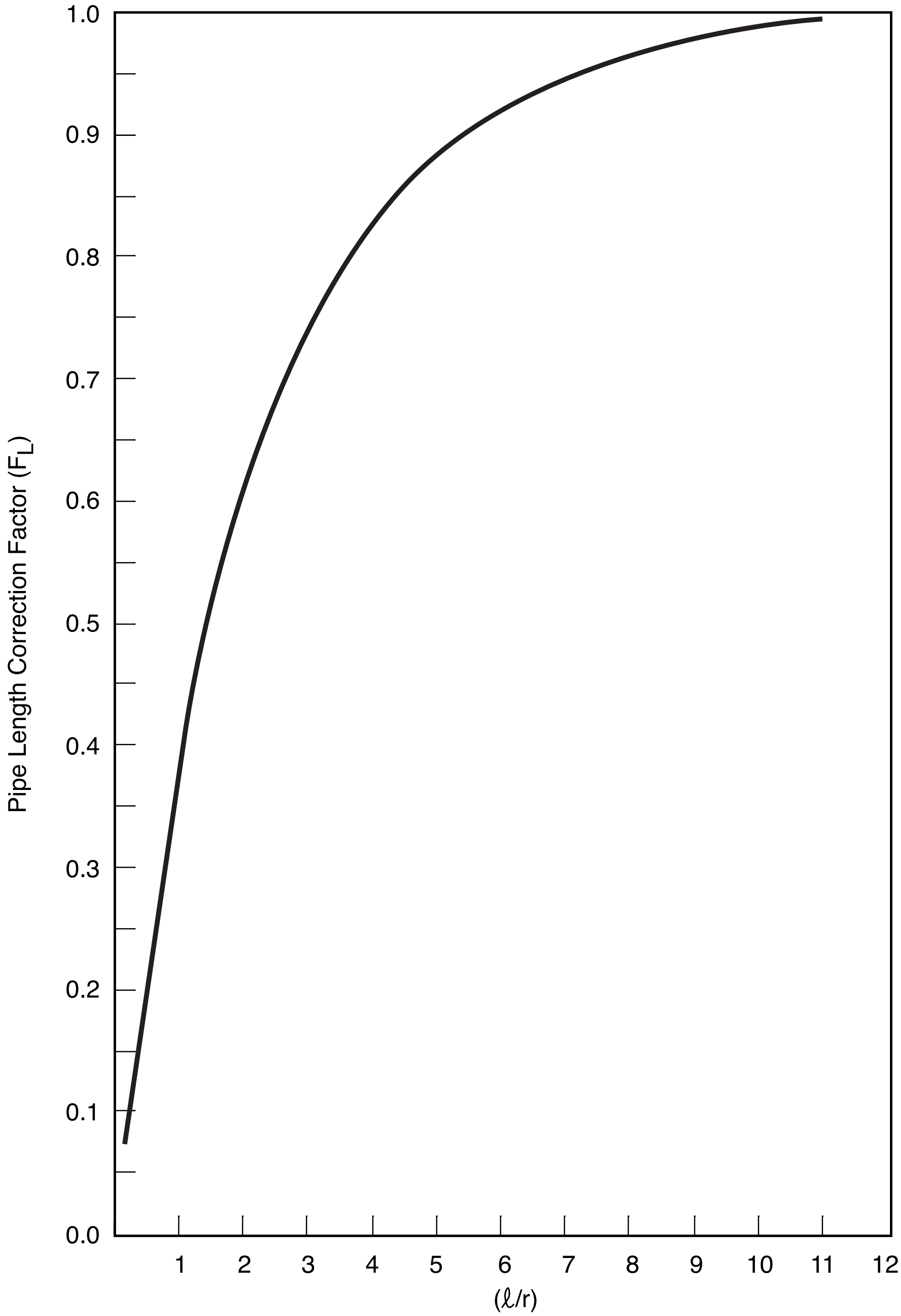


Pipe Diameter Correction Factor for Liquid Sources



Radial Distance Correction Factor for Gaseous Sources

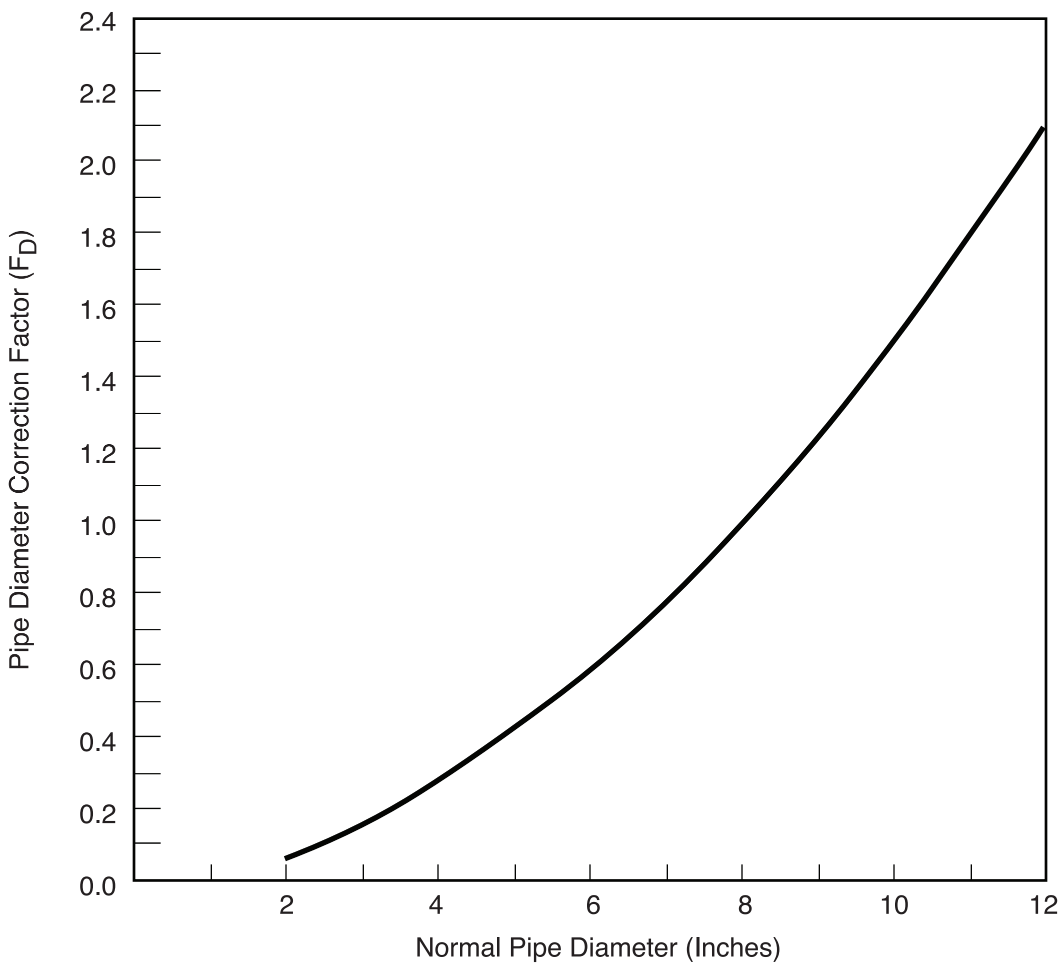
Draw. No. 970187.38 | Rev. | Figure J.B-15



Columbia Generating Station
Final Safety Analysis Report

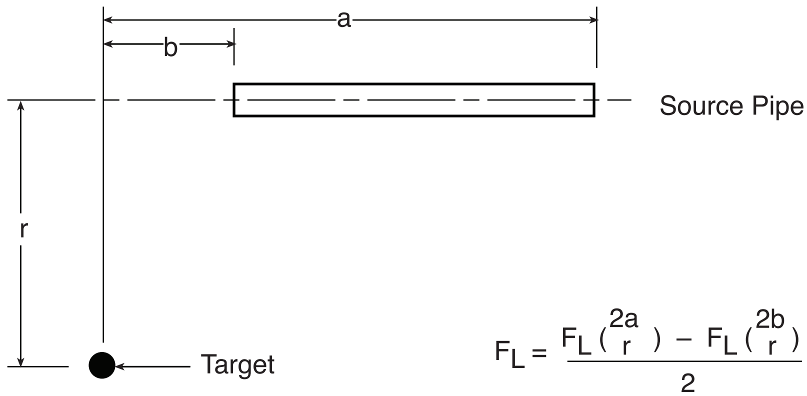
Pipe Length Correction Factor for Gaseous Sources

Draw. No. 970187.39 | Rev. | Figure J.B-16



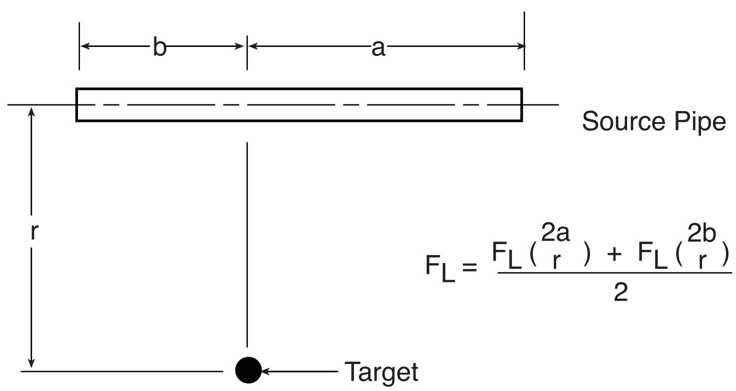
Columbia Generating Station **Final Safety Analysis Report**	**Pipe Diameter Correction Factor for Gaseous Sources**
	Draw. No. **970187.40** | Rev. | Figure **J.B-17**

Configuration 1



$$F_L = \frac{F_L\left(\frac{2a}{r}\right) - F_L\left(\frac{2b}{r}\right)}{2}$$

Configuration 2



$$F_L = \frac{F_L\left(\frac{2a}{r}\right) + F_L\left(\frac{2b}{r}\right)}{2}$$

Attachment J.C

PROCEDURE FOR THE CALCULATION OF SECONDARY CONTAINMENT RADIATION ZONE GAMMA DOSES

J.C.1 INTRODUCTION

Three Mile Island Lessons Learned Short Term Recommendations (NUREG-0578) Section 2.1.6.b, requires all nuclear power plant licensees to calculate post-loss-of-coolant accident (LOCA) environmental conditions for all safety-related equipment. This procedure is specifically concerned with the definition of the postaccident radiological environments in the secondary containment of Columbia Generating Station (CGS), a BWR.

The assumptions used in this procedure are based on a nonmechanistic LOCA scenario in which core damage is experienced at the beginning of the accident and primary containment isolation is achieved prior to radiation transport.

The radiation level at a given location inside the secondary containment of CGS during and following such an accident is defined by the following major source contributors.

Airborne gamma dose	Gamma ray dose from airborne radioactive sources inside secondary containment
Containment shine dose	Gamma ray dose from radioactive sources suspended in the drywell and the wetwell inside primary containment
Direct gamma dose	Gamma ray dose from piping containing recirculating radioactive fluids
Bioshield penetration streaming dose	Gamma ray dose from liquid piping and airborne radioactive sources inside primary containment which stream through bioshield wall penetrations into secondary containment

The methods presented in this procedure make it possible to calculate the worst-case gamma ray dose due to the above mentioned source of contributors inside radiation zones (see Section J.C.2 for the definition of radiation zones) of the secondary containment of CGS. The radiation zone dose calculated by using this procedure is applicable solely for the purpose of environmental qualification of safety-related equipment.

The following sections of this procedure describe the nomenclature, assumptions, and methods used in calculating radiation dose rates and cumulative doses. Section J.C.2 defines the terms and nomenclature found in this procedure. The assumptions and approximation used in developing the dose rate calculation method, as well as limitations to this method, are stated in Section J.C.3. Section J.C.4 provides a step-by-step procedure for determining the worst-case gamma dose rate and cumulative dose inside a particular radiation zone. The calculation of airborne beta dose is defined in a separate calculation procedure and is not included in this procedure (see Attachment J.E).

J.C.2 DEFINITION OF TERMS

This section contains the definition of the terms and symbols as used in this procedure:

CIND: (rads)
Cumulative integrated dose
Cumulative dose due to exposure to the decaying radioactive sources.

D_a: (rads/hr)
Airborne gamma dose rate
Gamma dose rate resulting from radioisotopes suspended in the atmosphere of the secondary containment.

D_d: (rads/hr)
Direct dose rate
Gamma dose rate resulting from the radioactive fluid contained inside recirculating pipes.

D_s: (rads/hr)
Shine dose rate
Gamma dose rate in the secondary containment resulting from radioisotopes suspended and deposited inside primary containment.

D_B: (rads/hr)
Bioshield penetration streaming dose
Gamma dose rate contributed by the liquid piping and airborne radioactive sources inside primary containment which stream through the bioshield wall.

D_t: (rads/hr)
Total gamma dose rate
Gamma dose rate contributed by the sum of airborne, direct, and shine from penetrations into secondary containment.

$$D_t = D_a + D_d + D_s + D_B$$

GF:
Geometric factor
Scaling factor used to convert semi-infinite airborne gamma dose to finite dose inside enclosed air spaces.

$$D_a = \frac{D_{a,\infty}}{GF}$$

$$GF = \frac{1173}{V^{0.338}}$$ (Reference J.7-39)

F_L: Length conversion factor
A scaling factor dependent on the source pipe segment length and spatial orientation relative to a target (see Figure J.C-1 for the calculation of this factor). F_L is used to convert the standard dose to the dose emitted by a pipe segment of finite length.

F_D: Diameter conversion factor
A scaling factor dependent on the source pipe diameter. F_D is used to convert the standard dose to the dose emitted by a pipe of specified diameter.

F_R: Radial distance conversion factor
A scaling factor dependent on the radial distance of the target from the source piping. F_R is used to convert the standard dose to the dose at a target of specified radial distance from the source piping.

F_t: Total dose contribution correction factor
A scaling factor used to convert the standard dose to the dose at a target from a pipe segment of specified geometry and orientation.

$$F_t = F_D * F_R * F_L$$

F_s: Sum of dose contribution correction factor
A scaling factor used to convert the standard dose to the radiation zone dose due to all the significant pipe sources in the zone.

$$F_S = \sum_{i=1}^{n} F_{ti}$$

Radiation zone: A region in the secondary containment defined to be such that gamma radiation calculated in the zone bounds the magnitude of dose received by the pieces of safety-related equipment located in that zone.

Source term: The total radiated gamma energy associated with a specified quantity of radioactive material released from the reactor as the result of a postulated accident.

Special sources: Radioactive source of such geometry or concentration that cannot be approximated by pipe segments of diameters 2 in. through 24 in. and containing contaminated liquid of activity concentration established in Section J.C.3.1. This can be a heat exchanger, standby gas treatment filter, pump, etc.

Standard dose: Gamma dose at a target having a radial distance of 8 ft from the centerline of an infinitely long, 8-in.- diameter schedule 40 pipe.

Target: The point in space chosen to represent the location of an object for which a dose rate and/or cumulative dose is being calculated.

Worst case target: Location of the piece of safety-related equipment inside a radiation zone which will experience the highest gamma dose among all the pieces of safety-related equipment in that zone.

J.C.3 ASSUMPTIONS, APPROXIMATIONS, AND LIMITATIONS

J.C.3.1 Basic Assumptions to be Used in the Analysis

Gamma doses and dose rates inside radiation zones will be determined for four types of radioactive source distribution:

Major Source	Contributors
Airborne gamma dose	Isotopes suspended in the atmosphere of the secondary containment
Shine dose	Gamma irradiation from the primary containment
Direct dose	Direct gamma irradiation from the radioactive fluid contained inside recirculating pipes
Streaming dose	Gamma irradiation from liquid piping sources inside primary containment and primary containment atmosphere streaming through bioshield wall penetrations

The dose contributed by each of these sources is determined by the location of the equipment, the time dependent distribution of the source, and the effects of shielding.

The assumptions used in determining the initial distribution and leakage of radioactivity in the primary containment are as follows:

a. 100% of the noble gases and 50% of the halogens initially in the reactor core will be distributed homogeneously within the primary containment free volume immediately following the postulated accident. Plateout of 95% of the elemental iodines is allowed to occur in accordance with Reference J.7-34;

b. 50% of the halogens and 1% of the remaining fission products in the core will be mixed homogeneously with the primary containment liquid space instantaneously. The primary containment liquid space is defined as the sum of the suppression pool liquid and the reactor coolant system (RCS) liquid. Assumptions a and b are NRC-recommended assumptions for defining radioactivity release fractions for the qualification of safety-related equipment (Reference J.7-2) and are consistent with the accident analysis (Reference J.7-13);

c. The core fission product source term is defined as the total product generated in the core after 1000 days at a reactor power of 3556 MWt. This represents the maximum burnup level in the core prior to radioactivity release and is conservative; and

d. Primary containment leakage of 0.50% volume/day was considered and is consistent with the assumptions established in Reference J.7-13.

J.C.3.1.1 Assumptions Used in the Calculation of Airborne Dose Rate Inside Secondary Containment

a. Activity that leaks into the secondary containment is homogeneously mixed with the secondary containment atmosphere prior to its removal from the atmosphere by the standby gas treatment system (SGTS) exhaust fans. This is consistent with the NRC-recommended assumptions used for calculation of doses inside primary containment (Reference J.7-2);

b. The SGTS flow rate of 2430 scfm is assumed to be the flow rate of the effluent air and is based on one reactor building air change per day;

c. Air that leaks out of the primary containment flows directly into the secondary containment. Bypass leakage is not considered. This is conservative when considering dosage in the secondary containment, since it maximizes the buildup of radioactivity in the secondary containment; and

d. Geometric factors provide a good approximation to convert the semi-infinite cloud dose to a finite cloud dose and is based on the results presented in Reference J.7-28 and based on average gamma ray energy of 0.733 MeV. The effect of variation of this parameter due to difference in gamma ray energies have been proven to be negligible (see Attachment J.B for justification).

J.C.3.1.2 Assumptions Used for the Calculation of Shine or Streaming Dose From Primary Containment

a. No depletion of activity due to leakage is assumed to maximize the source activity and is conservative;

b. The airborne source is assumed to be uniformly distributed in the drywell and in the wetwell air space. The effect of the plateout of iodine is not considered in secondary containment;

c. Activity in the wetwell water volume is assumed to be uniformly distributed in the sump water. Assumptions b and c are based on the plateout modeling and source term assumption contained within References J.7-2 and J.7-34;

d. The dosage at a point inside the region closest to the source is considered to be representative of the gamma dose in the region which maximizes the gamma ray dose at the region and is conservative; and

e. The liquid piping sources inside primary containment are assumed to be uniformly distributed in the RCS for the first 17 hr post-LOCA. The liquid piping sources inside primary containment are assumed to be uniformly distributed in the RCS plus the suppression pool after the first 17 hr post-LOCA. This is consistent with the CGS operations procedure to depressurize and utilize the alternate shutdown cooling mode within 17 hr post-LOCA once a degraded core condition is identified.

J.C.3.1.3 Assumptions and Approximations Used in the Calculation of Direct Doses

a. No valve leakage is assumed, which is consistent with Reference J.7-5, Item II.B.2, Clarification (2);

b. Schedule 40 piping is assumed, which is a conservative simplification of the calculation process. Because the majority of the pipe segments considered are schedule 40 piping, and because increases in pipe schedule can only decrease the dose rate at the targets, this approximation is considered to be conservative and appropriate;

c. Heat exchangers and pumps can be approximated as pipe systems. The volume of radioactive liquid in the component and its length are used to determine an equivalent volume of liquid. This is a crude approximation for dose rates contributed by complex geometries. Because the pump and heat exchanger walls are thicker than the pipe walls of schedule 40 piping, this assumption is conservative; and

d. Radioactive piping with diameters 2-1/2 in. or less was not modeled unless it was determined that such a pipe was a major source contributor. A major source contributor is defined as the only radioactive pipe in a target area or the radioactive pipe of closest proximity to the target. This is made because the dose contributions due to pipe segments of diameter less than 2-1/2 in. are generally negligible, unless they are major source contributors.

J.C.3.2 Limitations

The following limitations apply to the use of this procedure for the calculation of radiation zone doses.

a. This procedure is only applicable to the calculation of radiation zone gamma doses in the secondary containment of CGS;

b. The assumptions stated in Section J.C.3.1 are basic to the methodology used in this procedure. Changes in any of the assumptions will affect the accuracy of the results generated using this procedure;

c. The calculation of direct doses using the generic curves in this procedure is limited to liquid sources in schedule 40 pipe segments or equivalent pipe segments with nominal pipe diameters ranging from 2 in. to 24 in. Any deviation from these pipe geometries should be modeled as special cases. Note: Schedule 40 piping is used because the majority of the pipe segments to be considered are standard pipes (schedule 40). Increases in the pipe schedule only introduces conservatism in the results;

d. The results for direct dose calculated using the generic curves were found to be accurate to within 10% (see Reference J.7-39 for error study); and

e. Source piping located 40 ft or further from the target is generally an insignificant dose contributor. If its contribution is not found to be negligible, it should be considered as a special source.

J.C.4 PROCEDURES FOR THE CALCULATION OF SECONDARY CONTAINMENT RADIATION ZONE DOSES

This procedure describes the method used in calculating the gamma radiation doses inside radiation zones. For equipment located inside a zone, the following four sources contribute to the total dose level.

a. Airborne dose (gamma),
b. Direct gamma dose from sources within pipes,
c. Direct gamma shine dose from drywell and wetwell, and
d. Gamma streaming dose from drywell and wetwell.

A step-by-step procedure is discussed in the following sections for the calculation of the maximum total gamma dose and dose rates for each zone.

J.C.4.1 Procedure A: Radiation Zone Dose Calculation

The first step in preparing a zone dose calculation is to identify all the parameters to be used. This includes the identification of all the potential sources and targets, both inside and outside the zone, and the identification of the dimensions of the zone. Figure J.C-2 is a step-by-step flowchart of the calculation procedure. When identifying sources outside the zone, sources at the upper and lower elevations in the review process are included. A conservative dose estimate is used to determine whether a source outside a zone is a significant contributor. For example, if the closest pipe segment in the zone is a few feet away from a target, then the dose estimate will show that a pipe segment outside the room at 30 ft is insignificant by comparison. Conversely, if a target is located near a wall with several pipes on the other side of a wall, then those pipes may become significant source contributors and are included in the final evaluation for the target.

J.C.4.2 Procedure B: Airborne Dose Calculation in Secondary Containment

Because the semi-infinite airborne dose and dose rates are already calculated and shown in Figures J.C-6 and J.C-7, the only calculation involved in determining the airborne dose is the conversion of the semi-infinite cloud dose at reactor building concentrations to a finite cloud dose inside the cubicles in which the radiation zones are defined. The first step in this calculation is to determine the volume which defines the air space (or zone) of interest. An enclosed air space is defined as a cubicle, at least 95% shielded by concrete (or equivalent shielding) at least 1 ft thick.

To convert a semi-infinite cloud dose (calculated in Reference J.7-38) to a finite cloud dose, a geometric factor is used.

$$D_a(t) = \frac{D_{a,\infty}(t)}{GF} \tag{J.4-1}$$

where $GF = \frac{1173}{V^{0.338}}$ (Reference J.7-39) (J.4-2)

GF	=	geometric factor (dimensionless)
V	=	volume of the enclosed air space (ft^3)

Similarly,

$$CIND_a(t) = \frac{CIND_{a,\infty}(t)}{GF} \tag{J.4-3}$$

Figure J.C-3 is a step-by-step flowchart of the procedure for calculating airborne gamma doses.

J.C.4.3 Procedure C: Primary Containment Shine Dose Calculation

Containment shine doses are calculated using the QAD-P5A computer code. Guidelines for preparing input parameters are documented in Procedure E and Reference J.7-10. The modeling procedure and the accuracy of the results are highly dependent on the geometry to be modeled, specification of the source volume, and the selection of a buildup factor. Figure J.C-4 is a step-by-step procedure for calculating containment shine doses.

J.C.4.4 Procedure D: Direct Dose Calculation

The first step in the direct dose calculation (from Reference J.7-39) is the identification of the "worst-case" target. Normally, the worst-case target is the piece of equipment that is closest to the major source piping and can be selected by inspection. However, if situations arise such that the worst-case target cannot be chosen by simple inspection, order-of-magnitude calculations are performed for each potential worst-case target in the zone. These calculations are illustrated in Steps 3a through 3c of Figure J.C-5.

The next step is to identify special sources. Special sources are defined as source geometries that cannot be represented by liquid pipe segments between 2 and 24 in. in diameter. Example special sources are SGTS filters, reactor core isolation cooling (RCIC) steam pipe, turbines, and heat exchangers larger than 24-in. diameter. Other components such as pumps and small heat exchangers should be modeled as pipes. The pipe cross-sectional area is calculated by dividing the total fluid volume by the effective length of the component.

The contribution due to sources with shield walls is investigated next. Figure J.C-13 is used for this evaluation. If these sources are determined to be significant contributors, special QAD-P5A modeling procedures as described in Procedure E are followed.

It is unlikely that all sources under consideration will contribute significantly to the dose at a specific target. If all source contributions were to be calculated, the time involved in performing the calculation would be unnecessarily long without making a substantial improvement in the accuracy of the results.

Hence, as the sources are being identified, good judgment is used to distinguish between sources which contribute significantly to the target dose and those sources which do not.

An insignificant source is determined by comparing its dose contribution to the source making the largest dose contribution. The comparison is facilitated by arranging sources in decreasing order of importance and assigning rank numbers to the sources. The largest dose contributor is given a ranking number of 1. The largest dose contributor is determined by inspection of the sketches and drawings being used. The largest dose contributor is generally the longest segment with the largest pipe diameter and the least amount of intervening shielding between the target and source. All sources which are in the radiation zone and have been assumed to be insignificant contributors are listed as such to indicate that those sources have been considered.

<u>Equations Used in the Calculation of Dose Rates</u>

The following procedure is followed for the calculation of correction of dose rates factors of dose rates (Step 9 through Step 12 of Figure J.C-5):

a. Identify the radial distance of the pipe segment from the target; read F_R from Figure J.C-11.

   If the target is in contact with the source piping, read F_D from Table J.C-1 and set F_R and F_L equal to 1.

   (Note: dose rate is not a function of pipe length and radial distance.)

   If the target is geometrically in line with the source pipe segment, as shown in configuration 3 of Figure J.C-1, set $F_L=1$ and read F_D and F_R from Figures J.C-14 and J.C-15, respectively.

   (Note: F_L is defined here because dose rate is not sensitive to pipe length variation.)

b. Identify the pipe diameter; read F_D from Figure J.C-10.

c. Determine F_L from Figure J.C-12; use equations in Figure J.C-1 to calculate this factor.

d. The total dose contribution factor for a given pipe segment (I) is given as

$$F_t(I) = F_D(I) * F_R(I) * F_L(I)$$

e. When all the significant contributions have been calculated, sum the total dose contribution factors.

$$F_s = \sum_{n=1}^{n} F_t \ (I)$$

f. To determine if a source is negligible, the following test should be performed:

When N source segments are being considered and the dose contribution of ranking I is less than 1/10 of the dose rate calculated from the largest source divided by (N-I), the sources remaining should not contribute more than 10% to the total source contribution. This level of accuracy should be adequate for most calculations.

The total integrated direct dose and dose rate can be calculated.

$$D_D(t) = D_{Do}(t) \bullet F_s + D_D(t) \qquad \text{(Special Sources)}$$

$$CIND_D(t) = CIND_{Do}(t) \bullet F_s + D_D(t) \qquad \text{(Special Sources)}$$

where $D_{Do}(t)$ and $CIND_{Do}(t)$ are dose rates and cumulative doses for standard pipe segments and are found on Figures J.C-8 and J.C-9.

J.C.4.5 Procedure E: QAD-P5A Modeling Procedure

Direct dose contribution due to special sources and/or sources with shield walls should be calculated using the QAD-P5A computer code. This computer code is three-dimensional and calculates dose rates at specified target locations from radioactive volume, line, and point sources. Attenuation due to shield materials, if applicable, is also applied.

The accuracy of the results is highly affected by the manner by which the source volume is divided, and the position of the target relative to the source point. Therefore, a sensitivity study on the specification of the source volume should be performed. This can be achieved by increasing the number of source volume divisions until the dose rate results converge to within 5%.

Another factor to be considered is the specification of the buildup factor. As a general rule, aluminum buildup factor should be used when concrete shield is encountered, and iron energy buildup factor should be used when considering attenuation through steel shield.

J.C.4.6 Procedure F: Streaming Dose Calculation

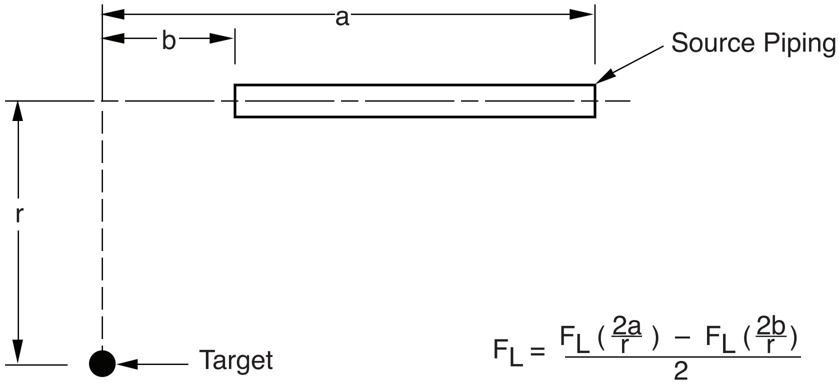
Containment streaming doses through the bioshield wall penetrations are calculated using the SCAP-BR and QAD-CG computer codes. The modeling procedure and the accuracy of the results are highly dependent on the geometry to be modeled, specification of the source volume, and the selection of a buildup factor.

Table J.C-1

Diameter Correction Factor (F_D) for Targets in Contact With the Source Piping

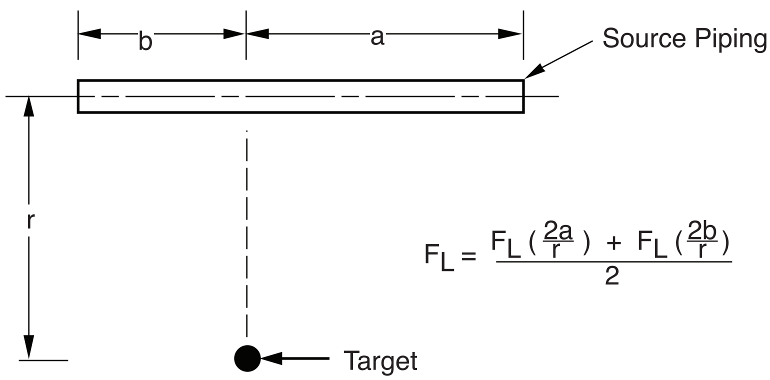
Nominal Pipe Diameter (in.)	Pipe Diameter Correction Factor (F_D)
2	18.4
4	24.4
6	54.6
8	33.3
10	35.3
12	35.3
14	35.5
16	33.7
20	32.0
24	29.6

Configuration 1



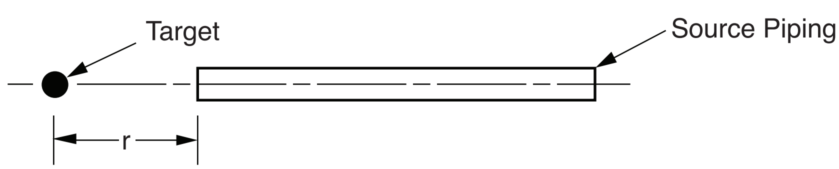
$$F_L = \frac{F_L\left(\frac{2a}{r}\right) - F_L\left(\frac{2b}{r}\right)}{2}$$

Configuration 2

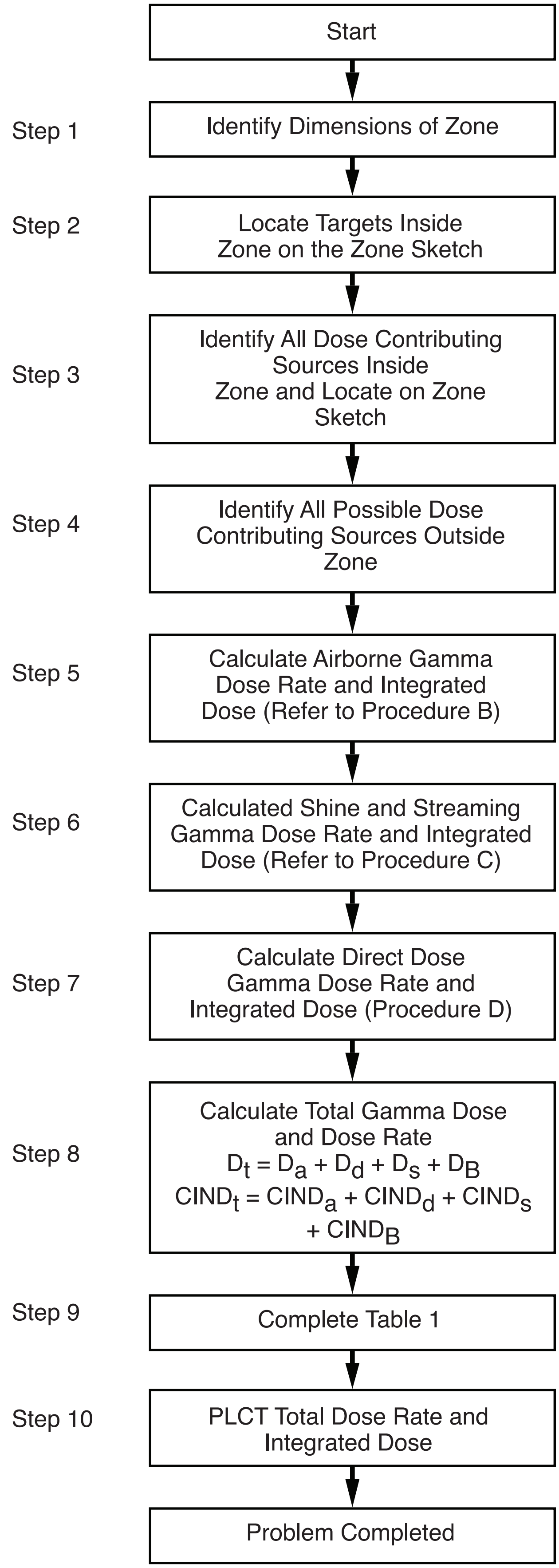


$$F_L = \frac{F_L\left(\frac{2a}{r}\right) + F_L\left(\frac{2b}{r}\right)}{2}$$

Configuration 3



$$F_L = 1$$



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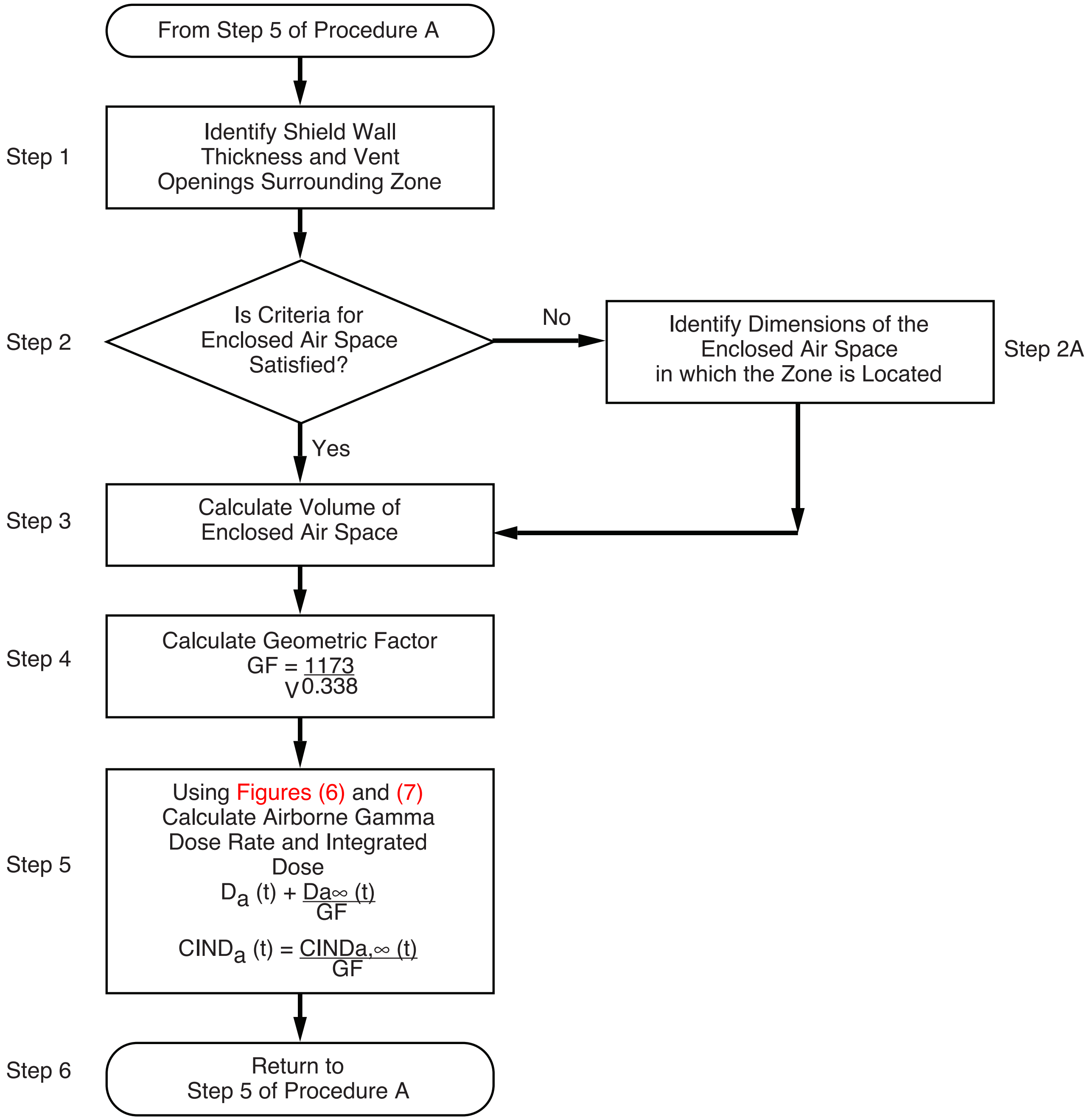
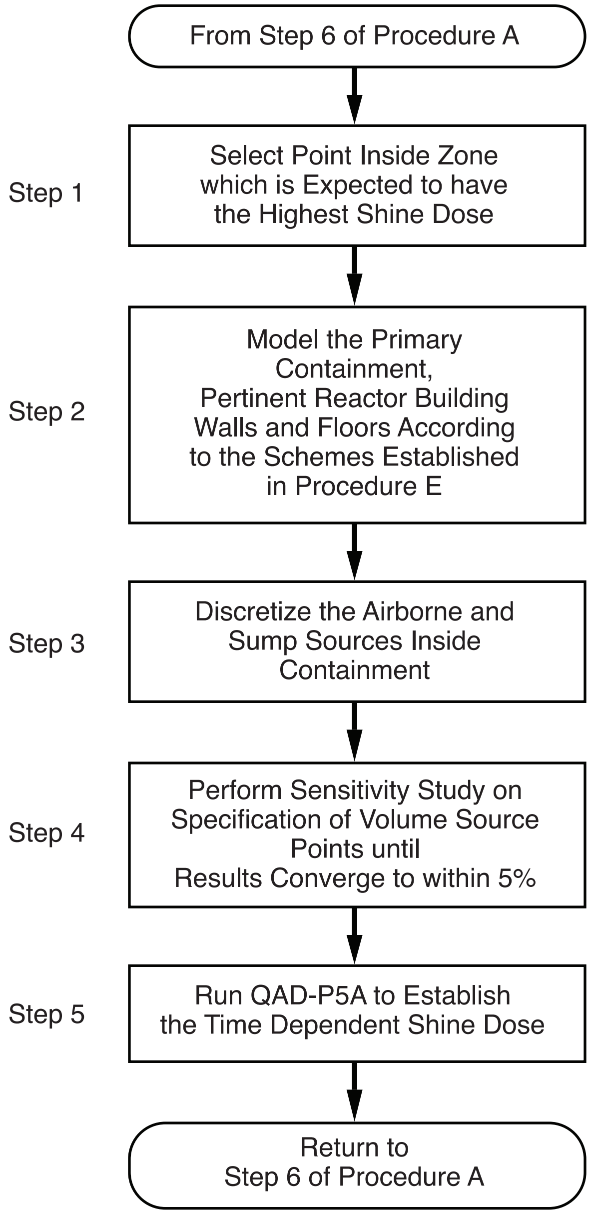
Procedure A: Procedure for Calculating Radiation Zone Doses

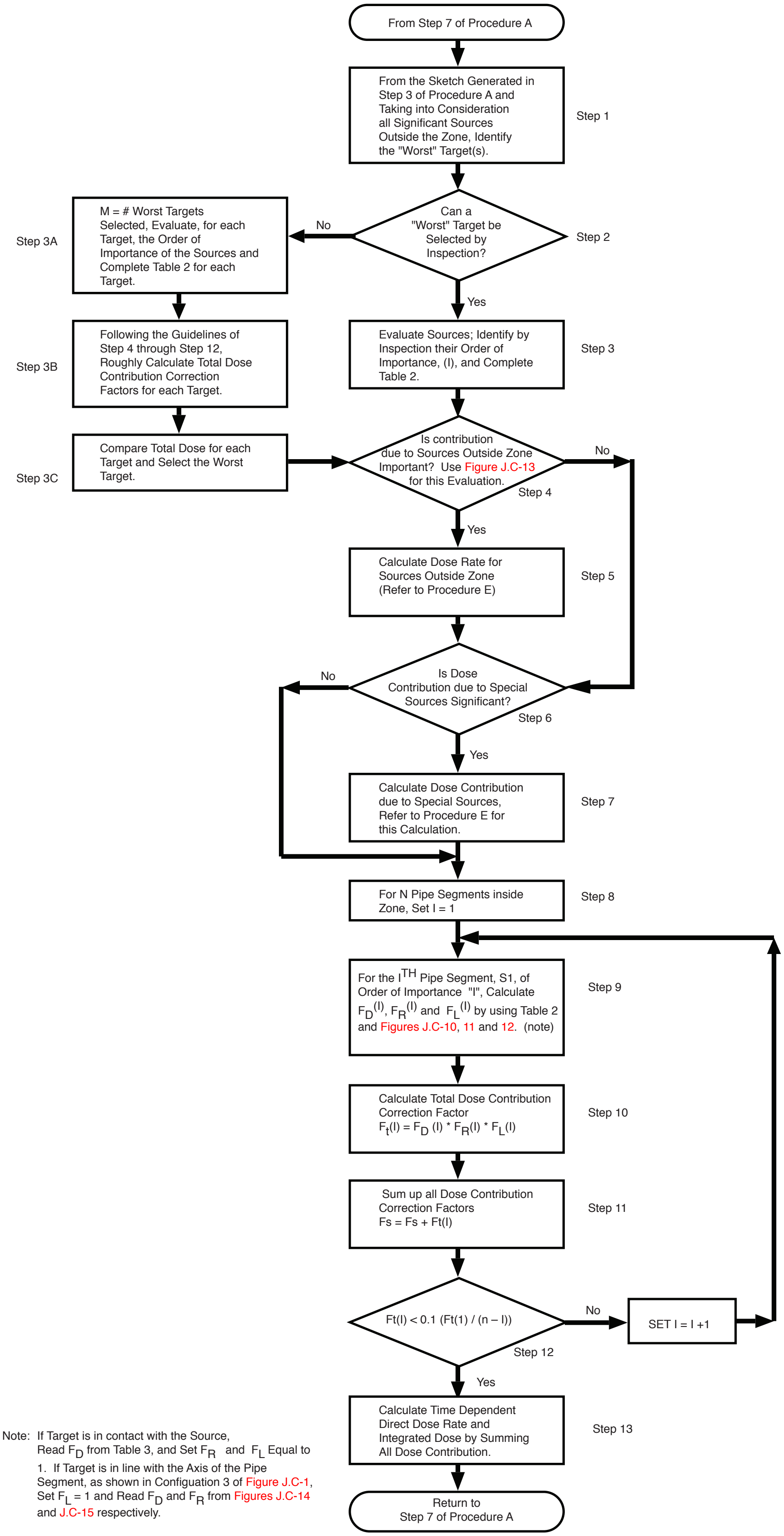
Draw. No. 970187.52 | Rev. | Figure J.C-2


From Step 5 of Procedure A
Step 1
Identify Shield Wall Thickness and Vent Openings Surrounding Zone
Step 2
Is Criteria for Enclosed Air Space Satisfied?
No
Identify Dimensions of the Enclosed Air Space in which the Zone is Located
Step 2A
Yes
Step 3
Calculate Volume of Enclosed Air Space
Step 4
Calculate Geometric Factor
$GF = \frac{1173}{V^{0.338}}$
Step 5
Using Figures (6) and (7) Calculate Airborne Gamma Dose Rate and Integrated Dose
$D_a(t) + \frac{Da_\infty(t)}{GF}$
$CIND_a(t) = \frac{CINDa,_\infty(t)}{GF}$
Step 6
Return to Step 5 of Procedure A



Procedure C: Procedure for Calculation of Containment Shine Dose

Draw. No. 970187.54 | Rev. | Figure J.C-4



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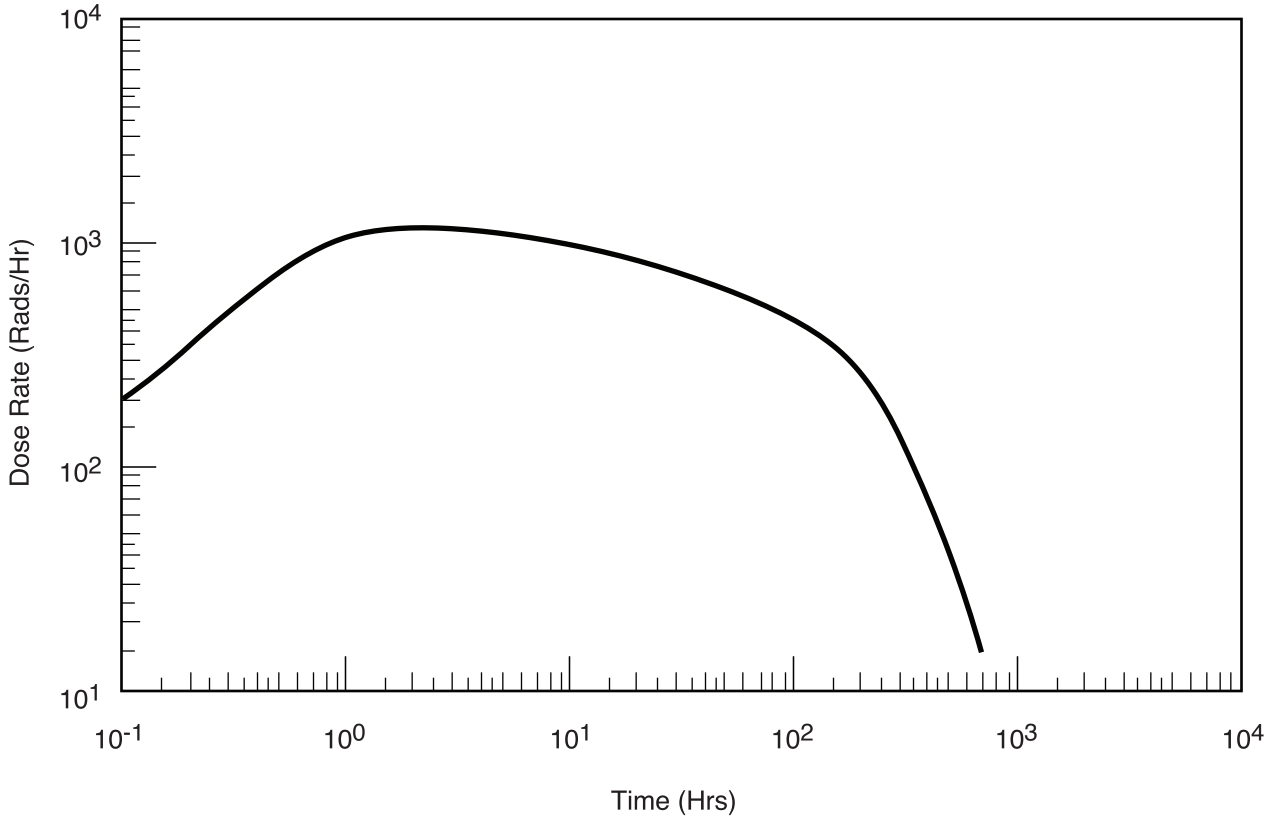
Procedure D: Procedure for Calculation of Direct Dose Rate and Integrated Dose

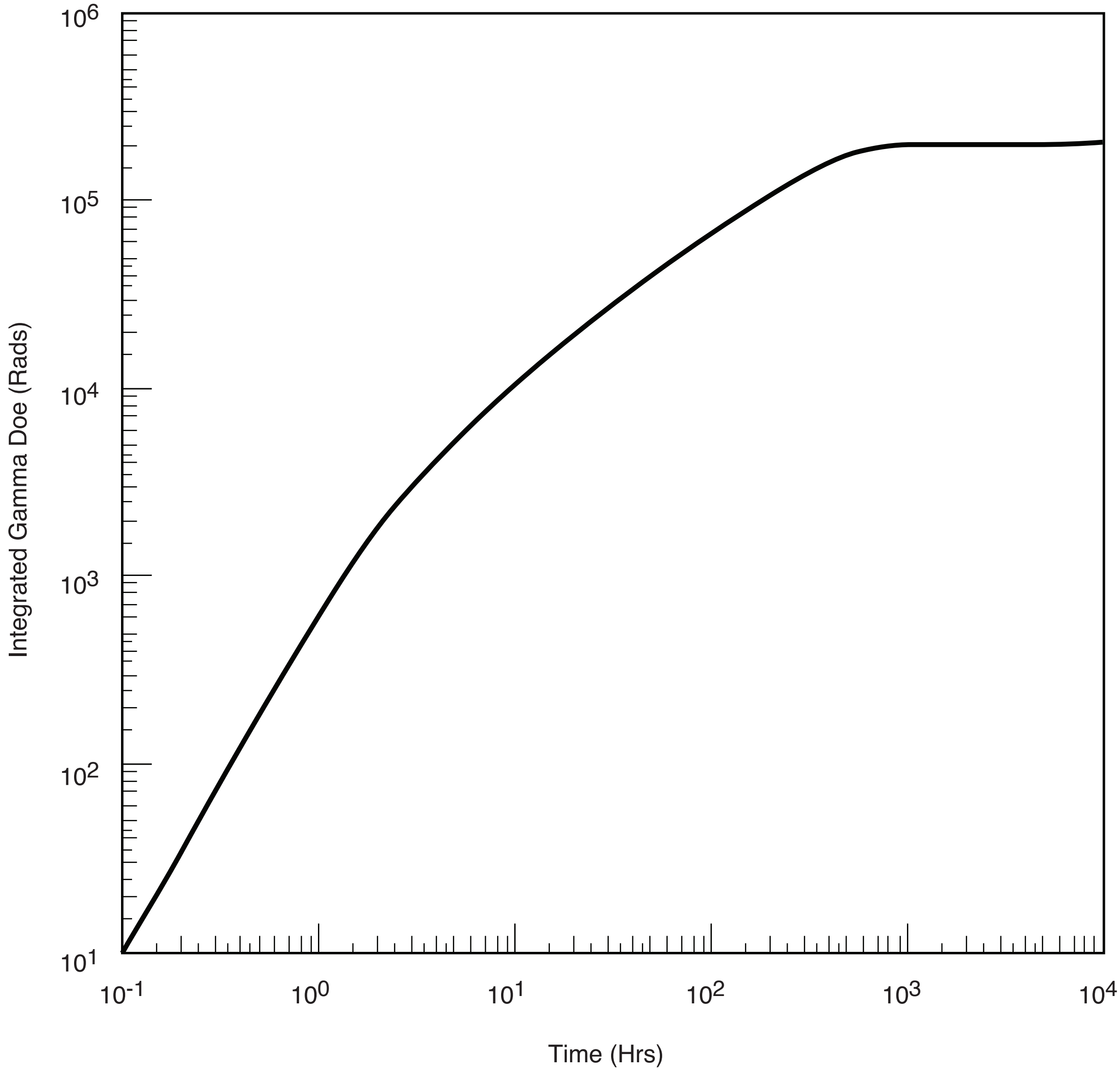
Draw. No. 970187.55 | Rev. | Figure J.C-5

Form No. 960690



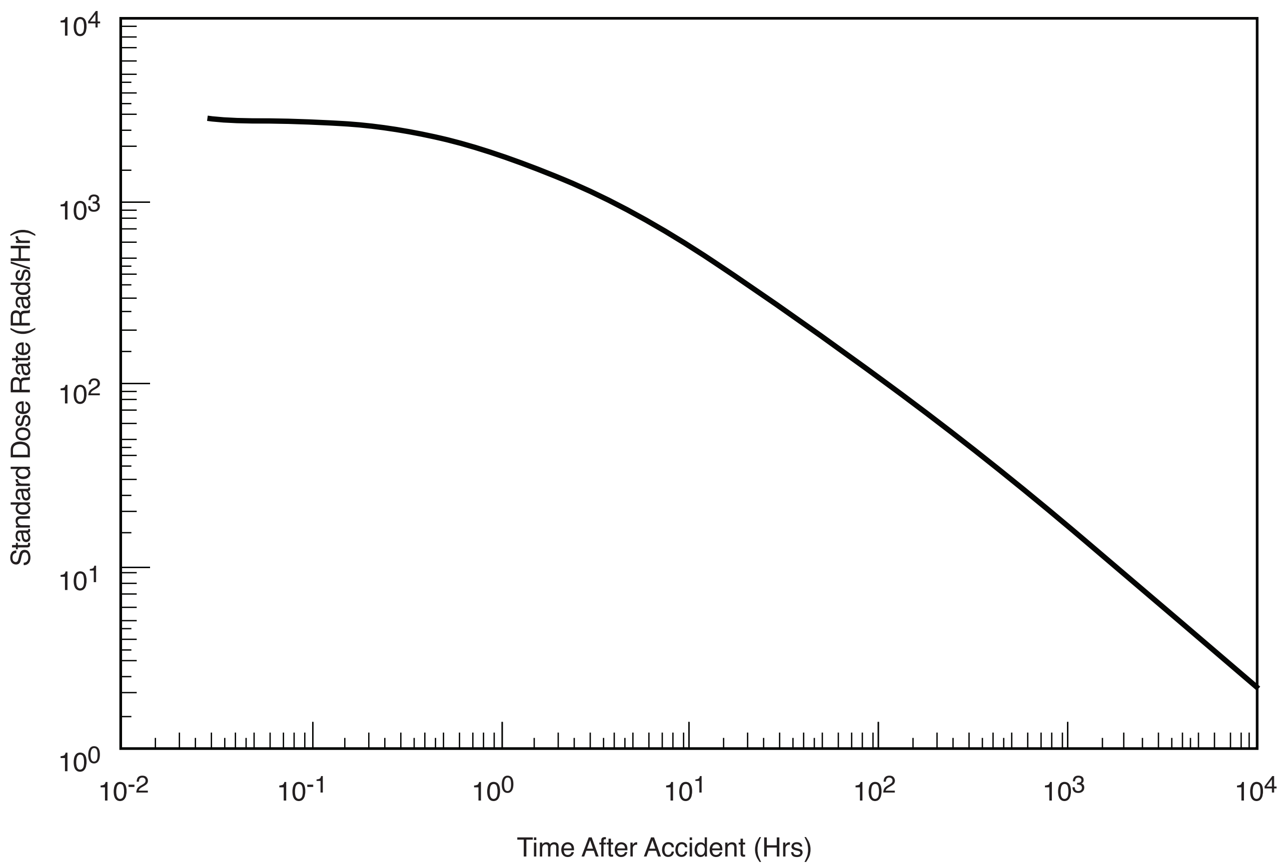
0.5%/Day Primary Containment Leakage Rate

Columbia Generating Station Final Safety Analysis Report	Time-Dependent Gamma Dose Rate for a Semi-Infinite Cloud of Fission Products at Secondary Containment Concentrations		
	Draw. No. 970187.56	Rev.	Figure J.C-6



0.5%/Day Primary Containment Leakage Rate

Columbia Generating Station Final Safety Analysis Report	Time-Dependent, Integrated Gamma Dose Rate for a Semi-Infinite Cloud of Fission Products at Secondary Containment Concentrations		
	Draw. No. 970187.57	Rev.	Figure J.C-7



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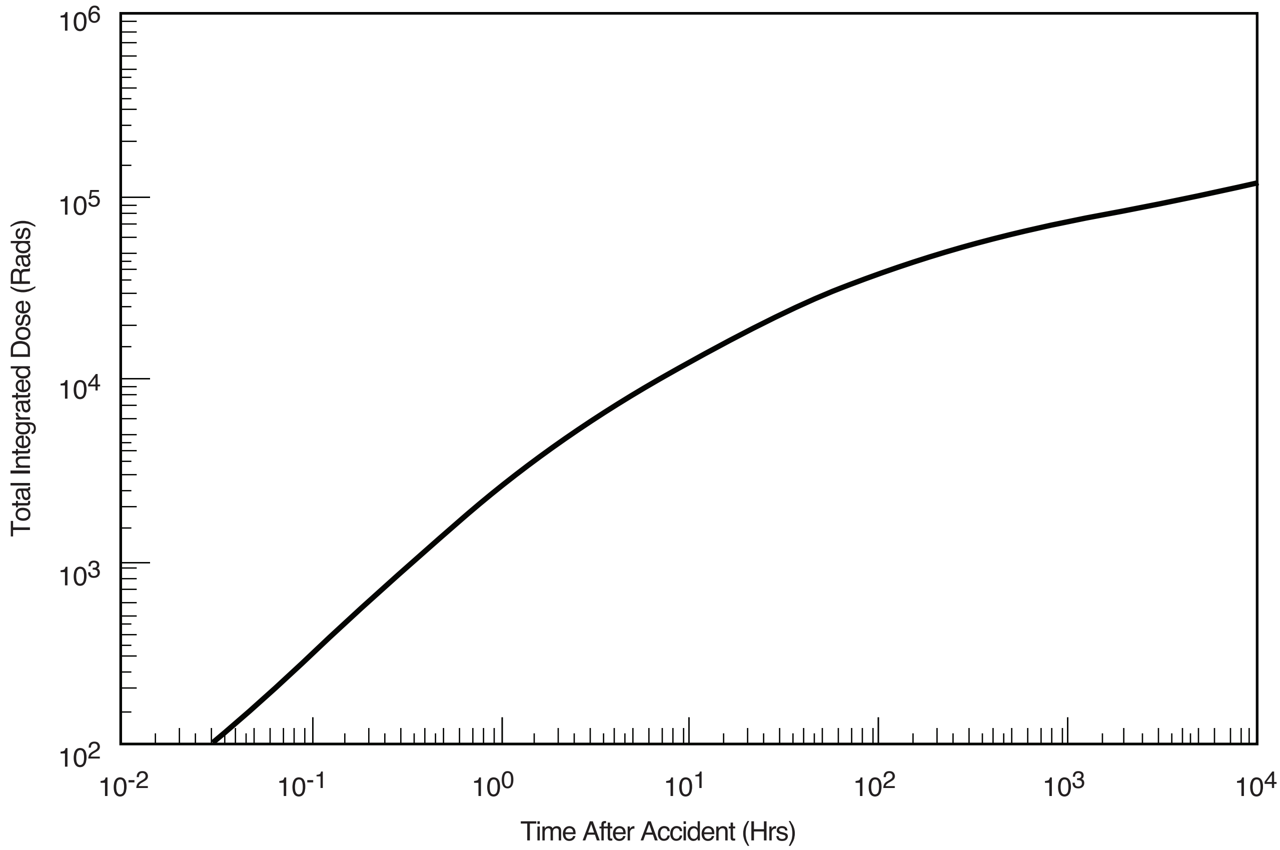
Gamma Dose Rate at Target 8 ft Away from Standard Pipe

Draw. No. 970187.58	Rev.	Figure J.C-8



Columbia Generating Station
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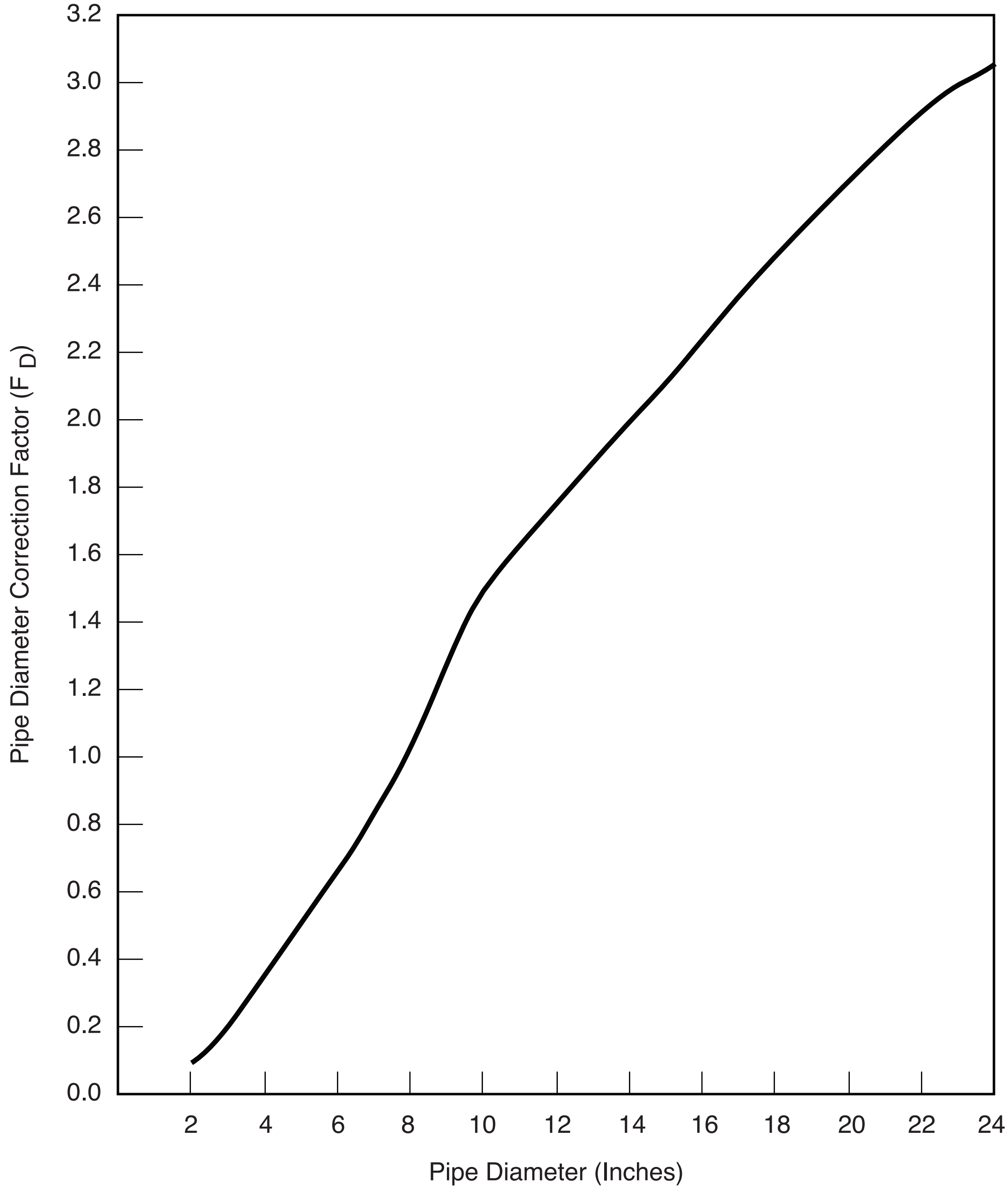
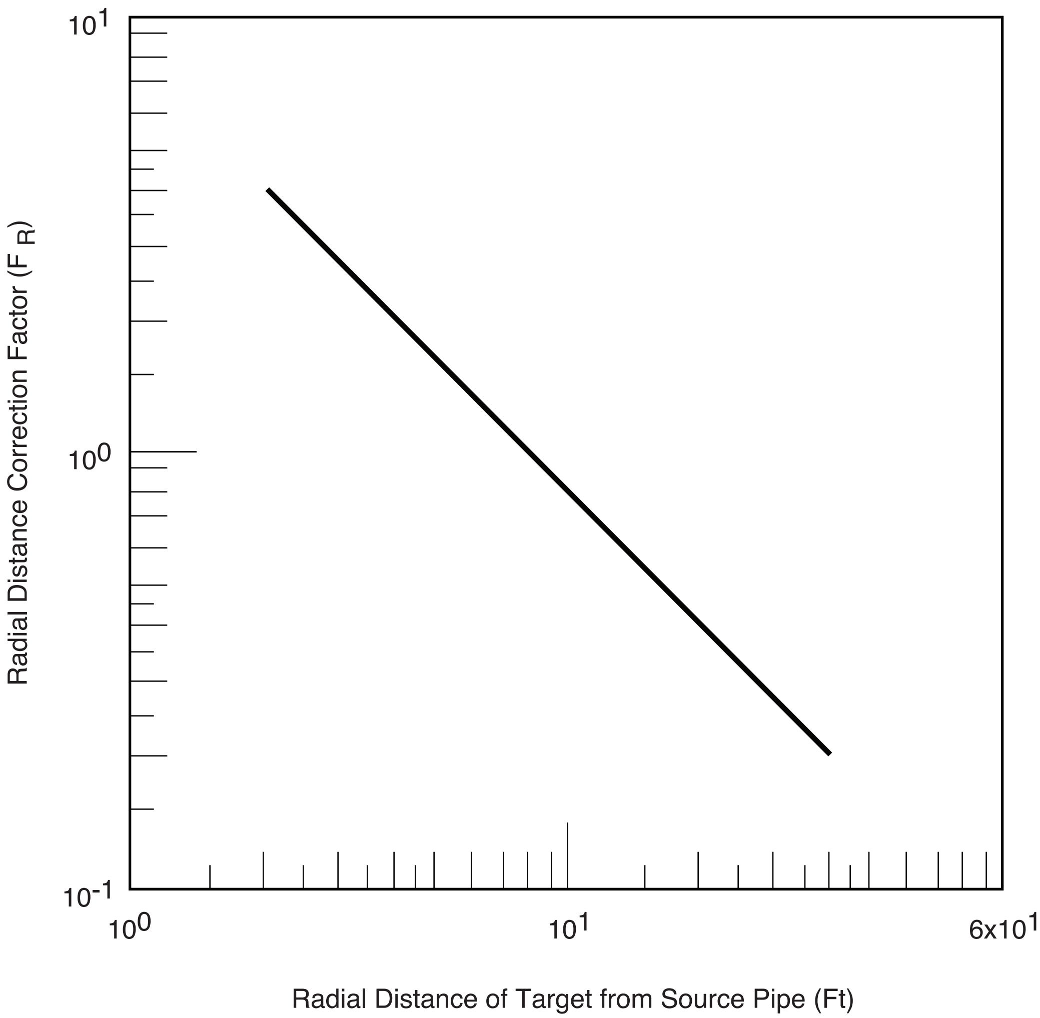
Gamma Integrated Dose at a Target 8 ft Away from Standard Pipe

Draw. No. 970187.59 | Rev. | Figure J.C-9

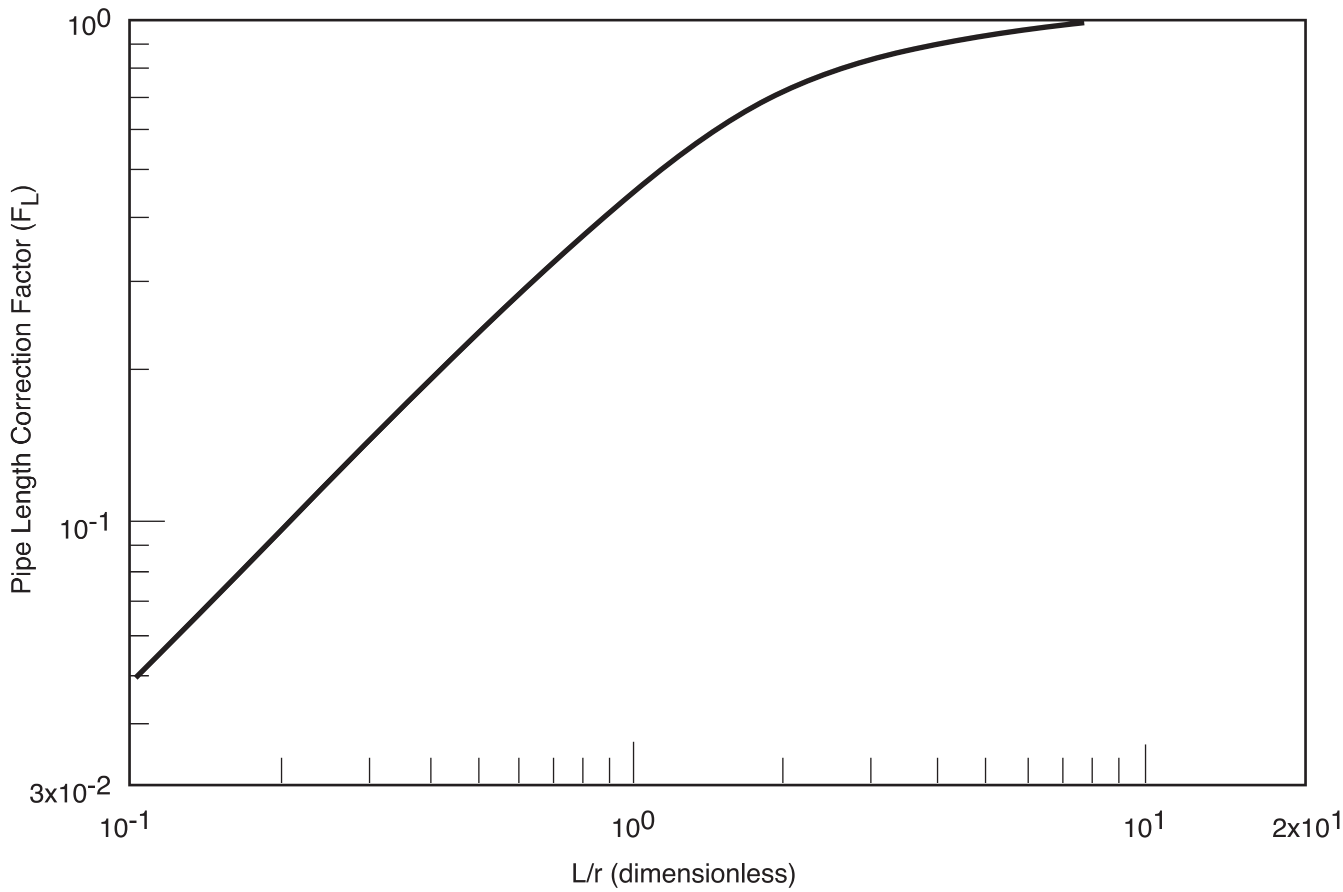


Columbia Generating Station Final Safety Analysis Report	**Pipe Diameter Correction Factor**
	Draw. No. 970187.60 | Rev. | Figure J.C-10



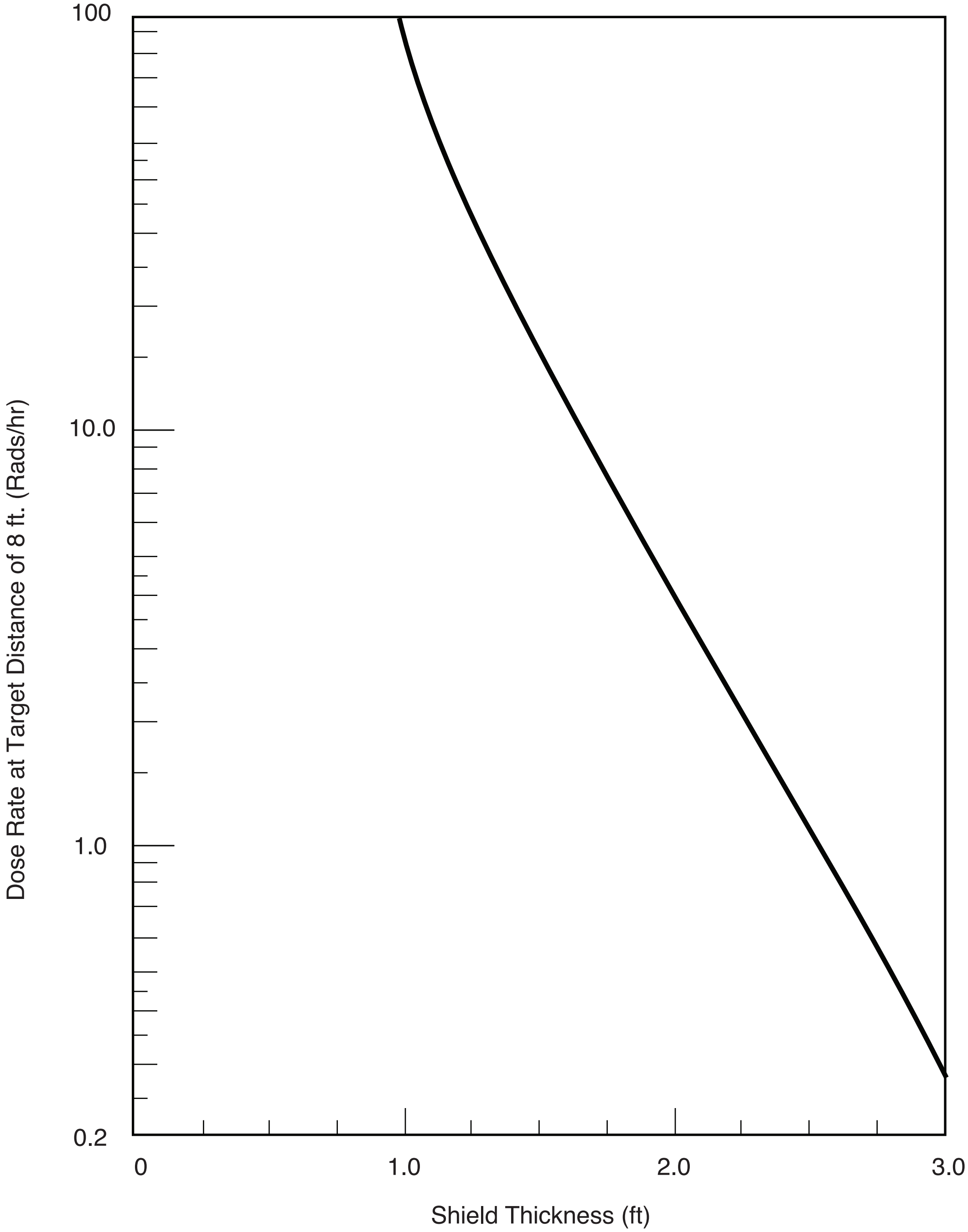
Radial Distance Correction Factor

Draw. No. 970187.61 | Rev. | Figure J.C-11



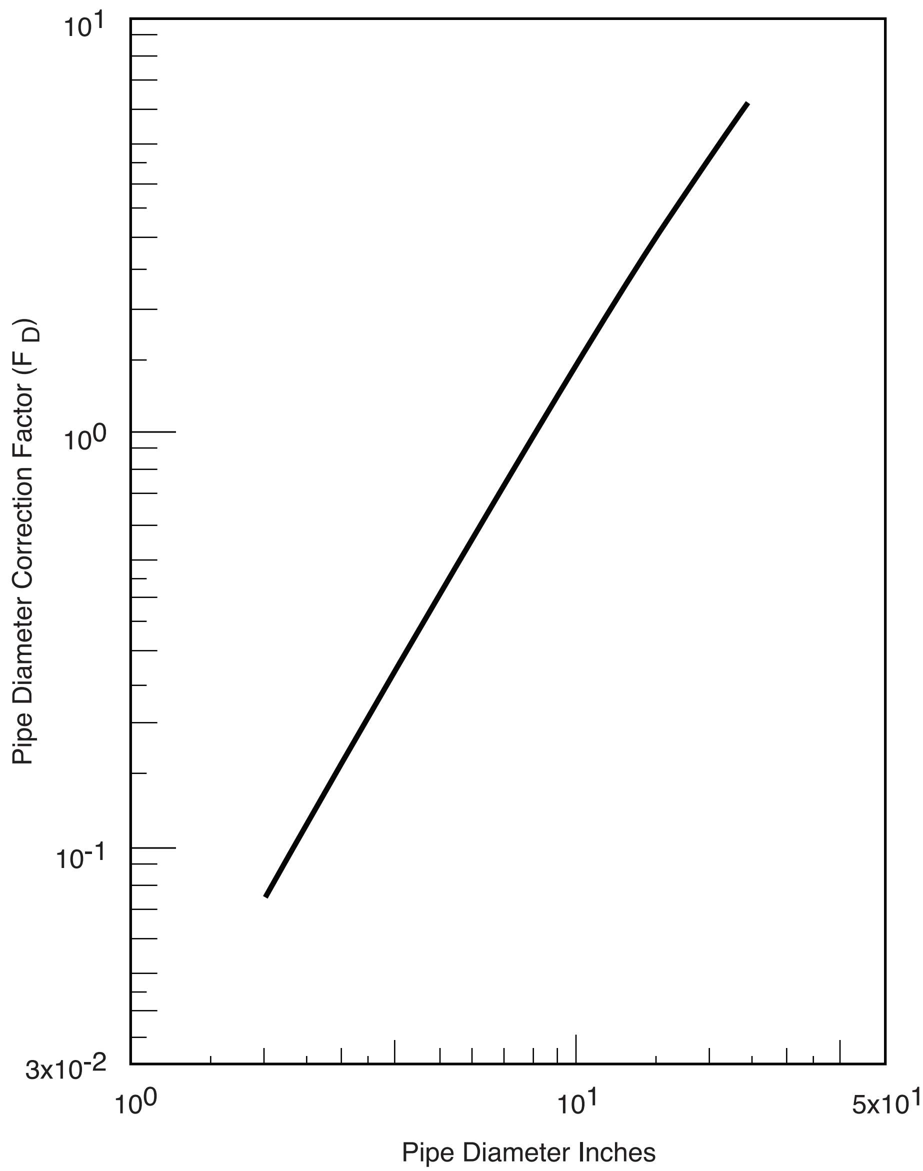
Pipe Length Correction Factor

Draw. No. 970187.62 | Rev. | Figure J.C-12



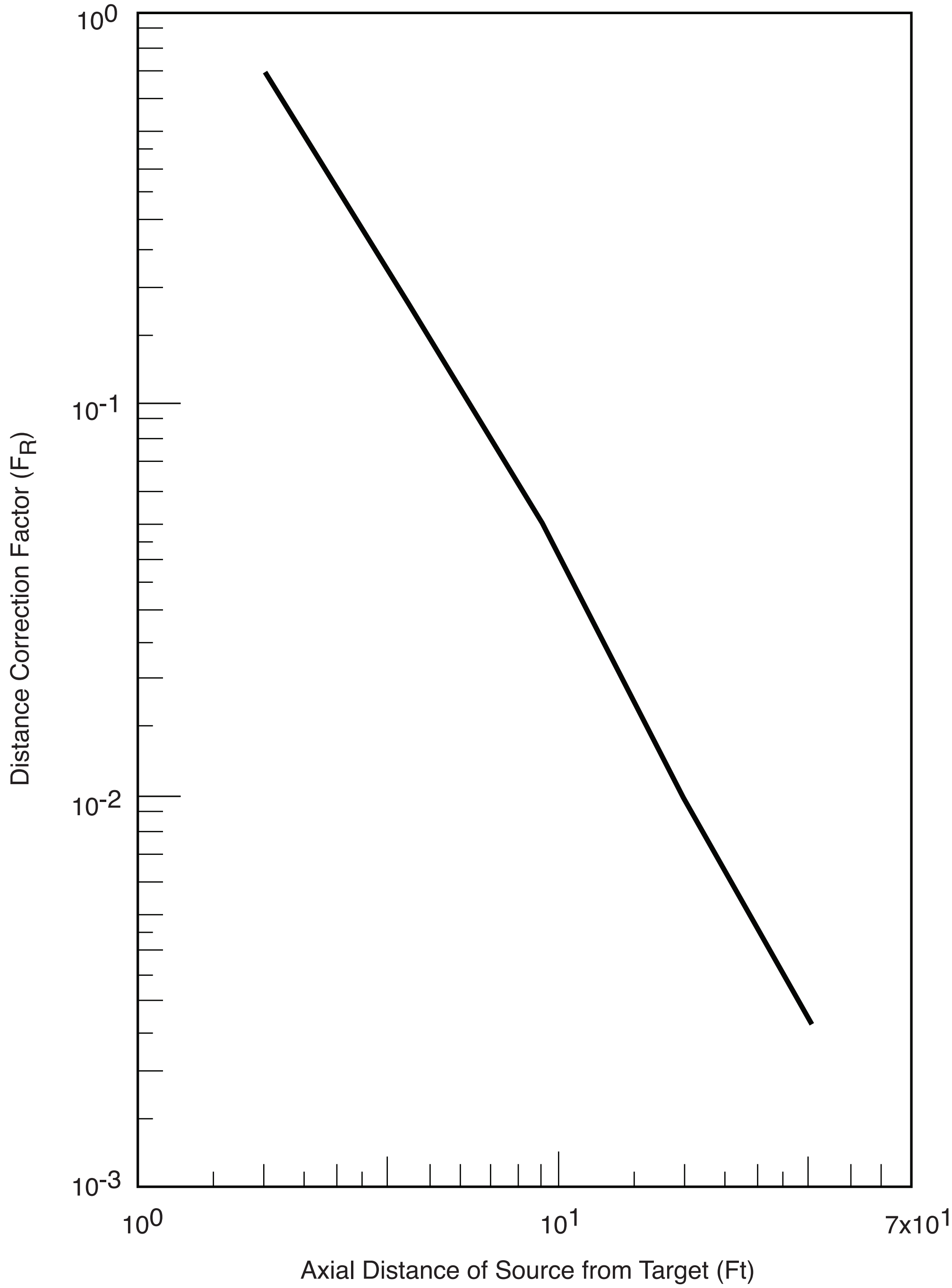
Note: This Figure is to be used for estimation purposes only.
Refer to Procedure E for calculating Dose rates outside shield walls

Columbia Generating Station Final Safety Analysis Report	Dose Rate Versus Concrete Shield-Thickness for Standard Pipe (8 in. Sch 40)		
	Draw. No. 970187.63	Rev.	Figure J.C-13



Configuration 3 of Figure J.C-1

Columbia Generating Station **Final Safety Analysis Report**	**Pipe Diameter Correction Factor for Targets Located Axially in Line with Source Piping**
	Draw. No. **970187.64** | Rev. | Figure **J.C-14**



Configuration 3 of Figure J.C-1

Columbia Generating Station
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Distance Correction Factor for Targets Located Axially in Line with Source Piping

Draw. No. 970187.65	Rev.	Figure J.C-15

Attachment J.D

CALCULATION OF THE RADIATION

The standby gas treatment system (SGTS) filters are located in the reactor building (el. 572 ft) and function to process the radioactive contaminated gaseous effluent from the primary and secondary containment. In the event of a loss-of-coolant accident (LOCA) in the primary containment the SGTS will be actuated. The gaseous contaminants that leak out of the primary containment will be filtered by the SGTS. It will adsorb the iodines in the charcoal filters and the particulates in the prefilters and high-efficiency particulate air (HEPA) filters. Plateout in the primary containment of the iodines released from the core was considered in the radiation assessment of the SGTS. Depending on the radioactive source distribution and the primary containment leakage rate, the radioactive iodine concentration in the filters will be increasing with time as more and more is deposited on the filters. Main steam isolation valve (MSIV) leakage is also considered in the radiation dose calculations.

The purpose of this study was to evaluate the time dependent gamma radiation level for safety-related equipment located near the SGTS filters and in adjacent rooms post-LOCA.

The time-dependent activity concentration in each of the filters is first calculated. The time and energy-dependent gamma activity levels on the SGTS filters is developed by a combination of computer runs and hand calculations and is used as input to the QAD-P5A computer code to calculate the gamma radiation levels for the pieces of safety-related equipment located in the room. A discussion of the analysis follows.

J.D.1 DESCRIPTION OF THE STANDBY GAS TREATMENT SYSTEM FILTERS

Figure J.D-1 is a drawing of the SGTS filter train. The SGTS consists of two fully redundant filter trains, each of which consists of the following components in series:

a. A demister to remove entrained water particles in the incoming air stream;

b. Two banks of electrical coil heaters designed to limit the humidity of the incoming air to 70% at design flow during post-LOCA conditions;

c. A bank of prefilters to remove large particles from the airstream (Figure J.D-2);

d. A bank of HEPA filters to remove the remaining particulates from the airstream (Figure J.D-2);

e. Two 4-in. deep beds of charcoal adsorber filters, arranged as shown in Figure J.D-1, are designed to capture the elemental and organic halogens from the airstream. The dimensions of the charcoal filters are shown in Figure J.D-3; and

f. A second bank of HEPA filters, identical to that described in item d above. The function of this second HEPA filter bank is to capture contaminated charcoal dust which may escape from the charcoal filters.

Both SGTS filter units are located in reactor building el. 572 and are automatically actuated and become fully operational within 34 sec of the event of any of the following three isolation signals:

a. High radiation in the reactor building ventilation exhaust duct,
b. High drywell pressure, and
c. Low water level in the reactor vessel.

J.D.2 CALCULATION OF TIME-DEPENDENT FILTER ACTIVITY CONCENTRATION

The analysis of the time-dependent transport of the radioactivity from the primary containment to the SGTS filters and the activity concentration on each filter is based on the following assumptions:

a. The SGTS filters are assumed to be loaded by iodine at a rate based on atmospheric leakage from primary containment of 0.67 wt %/day. This is composed of 0.5% direct from primary containment leakage and 0.17% via the MSIV. This is based on the primary containment rated leakage flow rate and the calculated MSIV leakage (Reference J.7-40). The containment rated leakage flow rate is 0.5%/day. The MSIV leakage was originally determined to be 0.23%/day, but a reevaluation has resulted in a revision of the MSIV leakage to 0.17%/day as referenced in J.7-40. Since the revision resulted in a lower value the original analysis with MSIV leakage of 0.23%/day is conservative. Thus, the radiation zone calculations were not revised to reflect the MSIV leakage of 0.17%/day since the original analysis was conservative;

b. Straight exhaust through the filters, with no mixing or holdup in the secondary containment atmosphere, is assumed based on an NRC recommended assumption for the analysis for fission product control systems (Reference J.7-41);

c. The elemental iodine in primary containment plateout on primary containment surfaces until one part in 200 of the elemental iodine remain airborne (0.5% of the total iodine). This is consistent with Reference J.7-14;

d. The released halogen fraction is 50% of the core inventory. This halogen fraction is assumed to be composed of 95.5% elemental, 2% organic, and 2.5% particulate iodines. This is consistent with Reference J.7-14;

e. The particulate halogens will be homogeneously distributed within the prefilters and the HEPA filters, while the elemental and organic iodines will be homogeneously distributed within the two charcoal filters of the filter train. This is conservative and necessary because the time-dependent collection of iodines in the filters has not been defined. The homogenous assumption is reasonable; and

f. Leakage past the MSIVs discharges directly to the inlet of the operating SGTS filter unit. Therefore, it bypasses the secondary containment volume. This is conservative and necessary because the time dependent collection of iodines in the filters has not been defined. The homogenous assumption is reasonable.

The time- and energy-dependent gamma activity concentration in the SGTS filters was first investigated as discussed in Section J.5.3.3. This analysis was performed by a combination of computer analysis and hand calculations. The activity concentration of a halogen isotope inside a SGTS filter is changing with time due to the following three mechanisms:

a. Transport of activity from the primary containment and deposition of the filters due to air leakage,

b. Depletion of activity due to radioactive decay and plateout of elemental halogens inside primary containment, and

c. Increases in activity levels due to daughter fission product generation from radioactive decay of other isotopes.

The activity balance on the SGTS filters can be described by (from equation J.B-16, Attachment J.B)

$$\frac{d}{dt}(A_i) = \underset{\text{leakage in}}{Q_1 C_{1i}(t)} - \underset{\text{decay}}{{}_i A_i} + \underset{\text{growth}}{{}_j\ {}_j A_j} \qquad \text{(J.D-1)}$$

where

A_i = activity (iodine) deposited on the SGT filters
$C_{1i}(t)$ = airborne concentration of iodine isotope "i"
Q_1 = flow rate (volume) from the primary containment

As in Attachment J.B (equation J.B-1, J.B-2) the growth term is negligible.

$C_{1i}(t)$ is given by equation J.B-8 of Attachment J.B as

$$C_{1i}(t) = (S_{iH}(t)/V_1)\ f_H(t)\ \exp(-Q_1 t/V_1) \qquad \text{(J.D-2)}$$

V_1 is the volume of primary containment

$f_H(t)$ is defined by

$$f_H(t) = f_e\ e^{-\lambda_p t} + f_p + f_o \quad \text{where } t \leq t_p \qquad \text{(J.D-3)}$$

$$f_H(t) = \left(\frac{f_e}{200}\right) + f_p + f_o \quad \text{where } t \geq t_p$$

Integrating (J.D-2) gives the following, where B is a constant to be determined:

$$A_i(t) = Be^{-\lambda_i t} + e^{-\lambda_i t} \int e^{\lambda_i t}\ Q_1 C_{1i}(t)\ dt \qquad \text{(J.D-4)}$$

$C_{1i}(t)$ is substituted into (J.D-4) from (J.D-2) to give

$$A_i(t) = Be^{-\lambda_i t} + \frac{e^{-\lambda_i t}}{V_1} \int Q_1\ S_{iH}(t)\ f_H(t)\ e^{(\lambda_i - Q_1/V_1)t}\, dt \qquad \text{(J.D-5)}$$

Substituting the definition of $S_{iH}(t)$ from equation J.B-5 of Attachment J.B, where $A_{1i}(0)$ is the original activity in primary containment

$$(S_{iH}(t)) = C_{1i}(0)\ e^{-\lambda_i t}\ V_1 = A_{1i}(0)\ e^{-\lambda_i t}$$

$$A_i(t) = Be^{-\lambda_i t} + A_{1i}(0)\ e^{-\lambda_i t} \int \frac{Q_1}{V_1}\ f_H(t)\ \exp\left[-(Q_1/V_1)^t\right] dt \qquad \text{(J.D-6)}$$

$f_H(t)$ consists of three chemical species: organic, particulate, and elemental iodine. Equation (J.D-6) must be solved for each species, so the species will be separated at this point:

$$\theta_o(t) = f_o$$

$$\theta_p(t) = f_p \qquad \text{(J.D-7)}$$

$$\theta_e(t) = f_e\ e^{-\lambda_p t} \quad 0 \le t \le tp$$

$$\theta_e(t) = \frac{f_e}{200} \quad \text{where } t_p \le t$$

$$f_H(t) = \theta_o(t) + \theta_p(t) + \theta_e(t)$$

To clarify the solution of (J.D-6), the following definitions are made:

$$X = {}^{-\lambda}p^{-q}$$

$$q = \frac{Q_1}{V_1}$$

Since $\theta_e(t)$ has step-function changes, solutions to (J.D-6) require a series solution - one for both of the time bands:

$$0 \le t_p \le \infty$$

<u>Organic Iodines</u>

Equation (J.D-6) for all times t becomes

$$A_i(t) = Be^{-\lambda_i t} + A_{li}(0)\ q\ f_o e^{-\lambda_i t}\ \frac{e^{-qt}}{-q} \qquad \text{(J.D-8)}$$

At t=0, A_i=0, so

$$A_i(t) = +\ f_o\ A_{1i}(0)\ e^{-\lambda_i t}\ \left[\ 1 - e^{-qt}\ \right] \qquad \text{(J.D-9)}$$

Since

$$A_{1i}(0)e^{-\lambda_i t} = S_{iH}(t)$$

from equation J.B-5 (from Attachment J.B), we define

$$A_i(t) = S_{iH}(t) \quad \phi_o(t) \qquad \text{(J.D-10)}$$

where

$$\phi_o(t) = + f_o\ (1 - e^{-qt})$$

$\phi_o(t)$ = fraction of organic halogens on the SGTS filters

Particulate Iodines

Particulate halogens are obtained in the same manner as organic halogens. The only difference is that f_o is replaced by f_p.

Elemental Iodines

For $0 \le t \le t_p$, equation (J.D-6) becomes

$$A_i(t) = Be^{-\lambda_i t} + A_{li}\ (0)\ e^{-\lambda_i t}\ q\ (e^{-\lambda_p t}\ f_e)\ \ e^{-qt}\ dt \qquad \text{(J.D-11)}$$

since at $t = 0$, $A_i = 0$

$$A_i(t) = q\ \frac{f_e}{x}\ A_{li}(0)\ e^{-\lambda_i t}\ (e^{xt} - 1) \qquad \text{(J.D-12)}$$

For $t_p \le t$, equation (J.D-6) becomes

$$A_i(t) = Be^{-\lambda_i t} + A_{li}\ (0)\ e^{-\lambda_i t} \int q\ (\frac{f_e}{200})\ e^{-qt}\ dt \qquad \text{(J.D-13)}$$

Integrating, with initial condition of $t = t_p$

$$A_i = \frac{qf_e}{x}\ A_{li}\ (0)e^{-\lambda_i t_p}\ (e^{xt_p} - 1)$$

$$A_i(t) = f_e\ A_{li}\ (0)\ \exp - \lambda_i t\ \frac{q}{x} \left[(e^{xt}[t p] - 1) + \frac{x}{200\,q}\ (e^{-qt}[t p] - e^{-qt}) \right] \qquad \text{(J.D-14)}$$

The activity on the SGTS filter may then be generally described by

$$A_i(t) = S_{iH}(t)\ \phi(t) \qquad \text{(J.D-15)}$$

where $\phi(t)$ is the fraction of released iodines located on the filters and is defined by

$$\phi(t) = \phi_o(t) + \phi_p(t) + \phi_e(t) \qquad \text{(J.D-16)}$$

where

$$\phi_o(t) = f_o\,(1 - e^{-qt})$$

$$\phi_p(t) = f_p\,(1 - e^{-qt})$$

$$e(t) = \begin{cases} F_e\,\dfrac{q}{x}\,(e^{xt} - 1) & (\text{For } 0 \le t \le t_p) \\ F_e\,\dfrac{q}{x}\left[(e^{xt}\,p_{-1}) + \dfrac{x}{200\,q}\,(e^{-qt}\,p_{-e}^{-qt}\;)\right] & (\text{For } t_p \le t) \end{cases}$$

J.D.3 CALCULATION OF RADIATION DOSE FROM THE STANDBY GAS TREATMENT SYSTEM FILTER

After the activity concentration in each filter segment is determined, the gamma radiation dose for safety-related equipment located in the SGTS filter room is determined by the use of computer code QAD-P5A (Reference J.7-10). The QAD-P5A modeling procedure as described in Attachment J.C is followed for this analysis. The following modeling assumptions were used:

a. Self-shielding of the filters is conservatively neglected because the density of the charcoal dust or the wire mesh (prefilter and HEPA filters) in the filters is low. Neglecting the self-shielding effect of the filters will not add too much conservatism to the results; and

b. Shielding due to the sheet metal filter housing is conservatively neglected due to computer code stability considerations. The shielding effect of the thin sheet metal filter housing is negligible.

One zone and four subzones were evaluated for the SGTS system and the five C1E/SRM components evaluated are

a. SGT-DV-1A3,
b. FPC-LIS-1A,
c. SGT-EHO-1B1,
d. SGT-MO-5B1, and
e. SGT-TE-6A1/7A1.

These targets are evaluated according to their proximity to the SGTS filters.

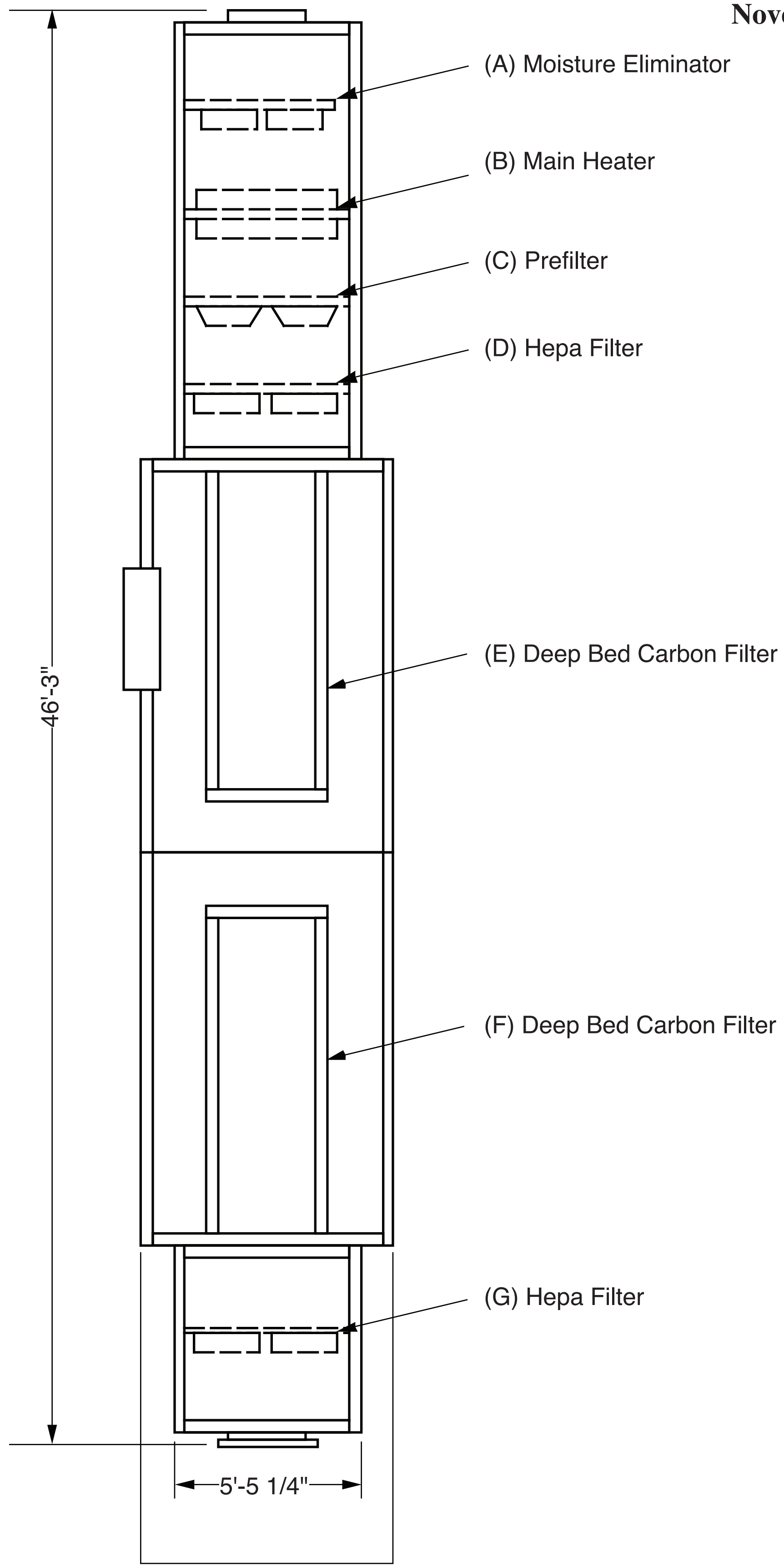
The time-dependent, gamma ray activity concentration as calculated using the method described in Section J.D.2 was used as input to the QAD-P5A model described in Attachment J.C. The dose rate results of this analysis were integrated numerically to give time-dependent, integrated doses. Table J.D-1 shows the direct gamma dose rate and integrated results for each of the five targets.

Table J.D-1

Direct Gamma Dose Rate and Integrated Dose Results
for Targets in the Standby Gas Treatment System Room

	FPC-LIS-1A			SGT-EHO-1B1			SGT-MO-5B1			SGT-TE-6Al/7A1	
Time (hr)	Dose Rate (rad/hr)	Integrated Dose (rad)		Dose Rate (rad/hr)	Integrated Dose (rad)		Dose Rate (rad/hr)	Integrated Dose (rad)		Dose Rate (rad/hr)	Integrated Dose (rad)
0	8.6E+02	4.3E+01		2.7E+02	1.4E+01		9.9E+02	5.0E+01		4.8E+04	2.5E+03
1	4.1E+03	2.3E+03		1.3E+03	7.3E+02		4.7E+03	2.6E+03		2.3E+05	1.4E+05
3	3.8E+03	1.0E+04		1.2E+03	3.3E+03		4.4E+03	1.2E+04		2.1E+05	5.8E+05
9	2.3E+03	2.9E+04		7.6E+02	9.3E+03		2.6E+03	3.3E+04		1.2E+05	1.6E+06
24	1.6E+03	5.8E+04		5.1E+02	1.9E+04		1.7E+03	6.6E+04		7.7E+04	3.1E+06
72	1.2E+03	1.2E+05		3.8E+02	4.0E+04		1.3E+03	1.4E+05		5.4E+04	6.3E+06
216	1.2E+03	3.0E+05		3.9E+02	9.5E+04		1.3E+03	3.3E+05		5.5E+04	1.4E+07
720	5.7E+02	7.5E+05		1.7E+02	2.4E+05		6.1E+02	8.1E+05		2.5E+04	3.4E+07
1440	7.6E+01	9.8E+05		2.3E+01	3.1E+05		8.1E+01	1.1E+06		3.3E+03	4.4E+07
2160	7.7E+00	1.0E+06		2.5E+00	3.2E+05		8.2E+00	1.1E+06		3.3E+02	4.6E+07
4320	1.0E-02	1.0E+06		6.4E-03	3.2E+05		1.0E-02	1.1E+06		2.3E-01	4.6E+07

Amendment 53
November 1998



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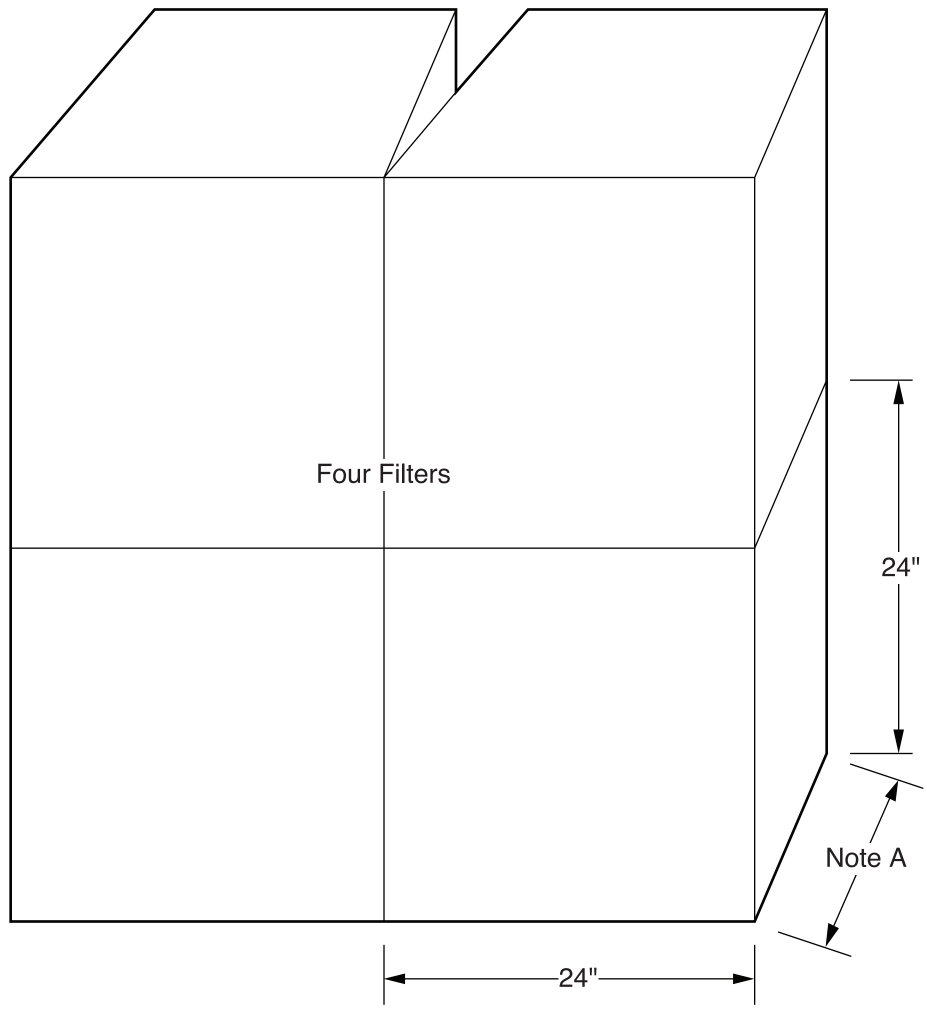
Standby Gas Treatment Filter

Draw. No. **970187.66** Rev. Figure **J.D-1**



Note A: Prefilter 8"
HEPA Filter 11 1/2 "

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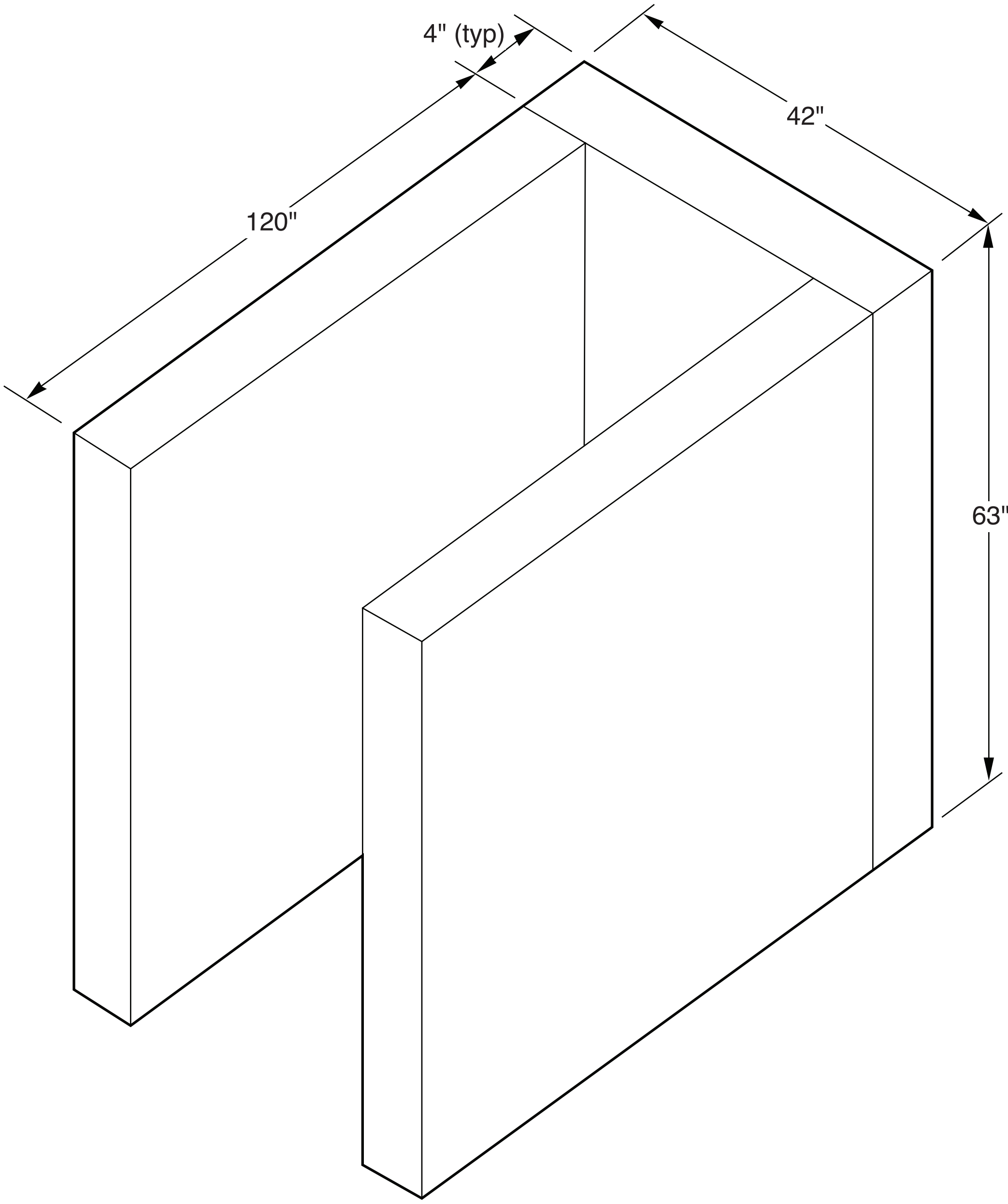
Geometry of Prefilters and HEPA Filters

Draw. No. 970187.67 | Rev. | Figure J.D-2

Form No. 960690



Not to Scale

Columbia Generating Station
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Geometry of Charcoal Filters

Draw. No. 970187.68 | Rev. | Figure J.D-3

Attachment J.E

BETA DOSE CALCULATION METHOD

The source volume used for the beta dose analysis in secondary containment is a sphere surrounded by a shell of sufficient thickness to stop all outside beta particles from entering the source volume. This spherical source volume is conservative for any generalized source volume shape. The dose at the center of the sphere is higher than the dose at any point of any generalized source of equal total volume.

The assumptions used in this analysis are as follows:

a. Atmosphere inside the equipment casing is identical to the atmosphere in the reactor building which is conservative because there will be some actual delay in transport of the gaseous fission products into the equipment;

b. The initial beta source term used was 100% of core noble gases and 50% of core halogens based on NUREG-0588, Revision 1 and NUREG-CR/0009 (References J.7-29 and J.7-34);

c. Daughter products of the airborne noble gases and halogens are included in the calculation of the airborne dose. This is conservative and was required by the use of ORIGEN2 as a source code (Reference J.7-8);

d. Plateout of halogens inside primary containment was utilized as allowed in accordance with Reference J.7-34. The dose contribution of fission products plated out on equipment casings was neglected. This is based on the NRC recommended assumptions (Reference J.7-34). The deletion of dose contributions from fission products plated out on equipment casings is acceptable, since equipment surface areas are small relative to the available containment surface area. In addition, the beta radiation emitted from plated out fission products would be absorbed in the equipment casing and, hence, would not affect internal components;

e. The primary to secondary leak rate is 0.5% of primary containment, wt %/day is consistent with the assumptions established in Reference J.7-2;

f. The standby gas treatment system (SGTS) operates at the minimum flow of 2430 scfm based on the SGTS flow rate assumption of one reactor building air change per day;

g. Primary to secondary leakage is homogeneously mixed in the secondary containment atmosphere consistent with the NRC-recommended assumptions used for the calculation of doses inside primary containment (Reference J.7-2);

h. No halogen plateout in the secondary containment was assumed; and

i. A spherical volume and equipment casing will be used which is conservative.

The beta dose to equipment is dependent on the internal volume size of the piece of equipment. The beta dose is determined through the use of any energy dependent geometry factor and a ratio of the internal equipment volume to an infinite cloud. The beta dose contribution is excluded from the total integrated radiation doses shown on the radiation zone maps and tables for the C1E* equipment in the reactor building. If determination of a beta dose contribution to a C1E* component is required then a calculation to determine the internal volume size and perhaps the angle of incidence of the beta cloud to the sensitive component is performed. The results of the beta calculation are then included in the equipment qualification files for that beta sensitive equipment.

The beta calculation is determined by the airborne dose at the center of the spherical source as a function of the volume of the sphere.

The variation of beta dose rate from a typical beta energy distribution in a one-dimensional absorbing medium can be approximated by the formula:

$$D(X) = A \exp(-\mu_E X) \quad \text{(J.E-1)}$$

where

D(X)	=	dose at a point X
A	=	constant
X	=	position in the material
μ_E	=	a parameter that depends on beta energy

This relationship holds approximately up to the point where all beta particles are absorbed. This point is called the range of the beta particles. The range of a beta particle is dependent upon the energy of the beta particle and is denoted r_E.

Both of the parameters μ_E and r_E may be determined by empirical formulas given below, based on the maximum energy of the beta particles, and approximately independent of the absorbing medium.

* Environmental qualification (EQ) of safety-related mechanical equipment has been eliminated from the overall Columbia Generating Station EQ program (SRM).

$\mu_E = 17\rho\ (E_{max})^{-1.14}$ (J.E-2)

$^rE = (0.412/\rho)\ E^n$ for $0.01 \leq E \leq 3$ (J.E-3)

$= (0.530E - 0.106)/\rho$ for $2.3 \leq E \leq 20$ (J.E-4)

ρ is material density (in g/cm^3)
E is energy of beta particle (in MeV)
μE is in cm^{-1}
r_E is in cm
n is 1.265 - .0954 LnE

The dose at a given point from a single beta source is now transformed into a dose from a uniform concentration of airborne sources which extend from radius zero to radius r. K is a constant.

$$D(r) = K(1 - \exp(\mu_E r)) \quad \text{(J.E-5)}$$

This relationship is valid for $r \leq r_E$. At $r \leq r_E$, none of the beta particles originating beyond r_E reach the target point. Hence, at this radius, an effective infinite medium for airborne beta radiation has been reached. The dose from a volume such that $r \geq r_E$ is equal to the dose from an infinite volume, which is denoted $D\infty$.

The dose as a function of volume radius is thus found to be given by the dual relation:

$$D(r) = D\infty \frac{(1 - \exp(-\mu_E r))}{(1 - \exp(-\mu_E r_E))} \qquad 0 \leq r \leq r_E \quad \text{(J.E-6)}$$

This relation may be transformed to a function of volume by noting that $V = 4\pi r^3/3$.

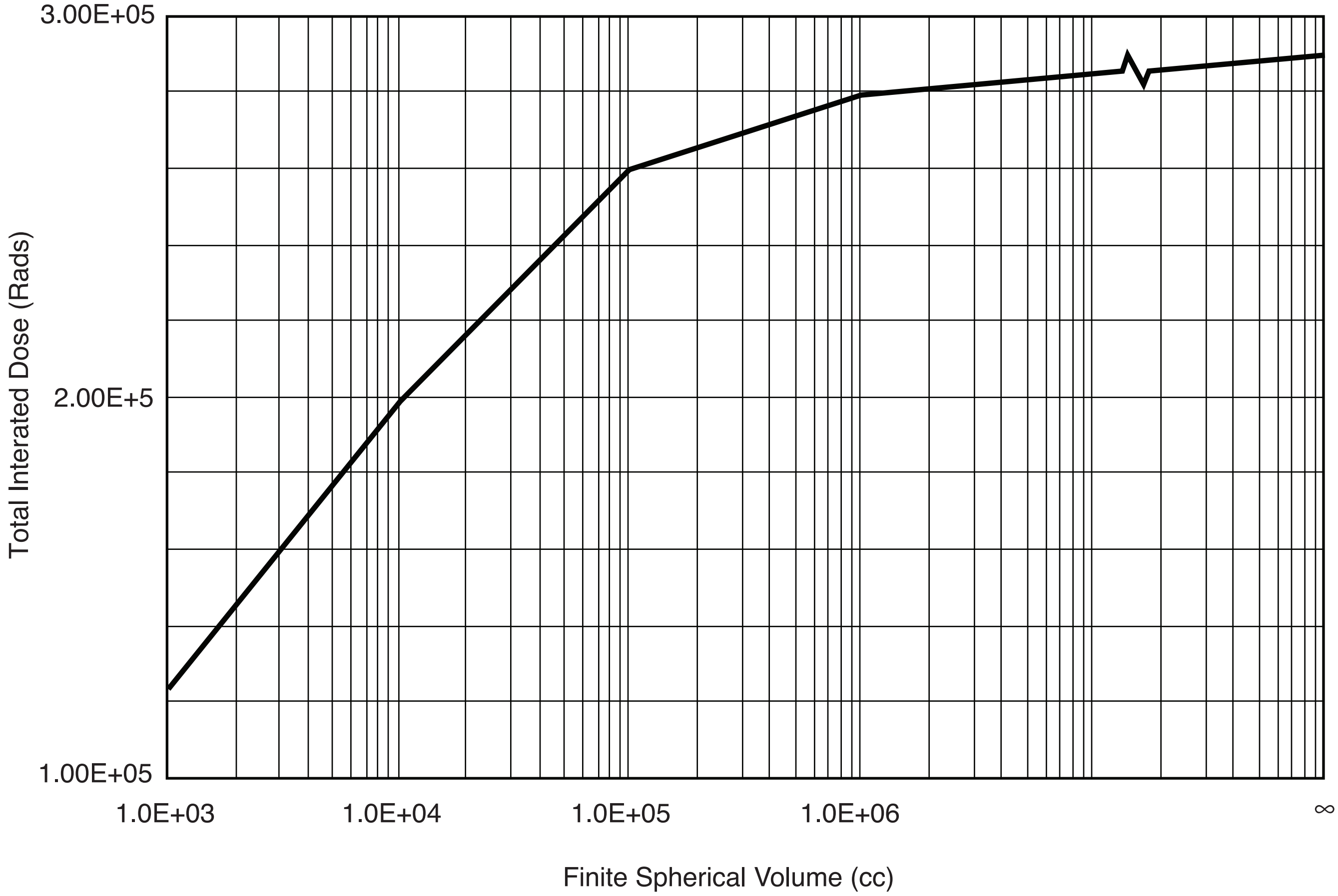
Since μ_E and r_E vary for each beta energy, this equation cannot be solved analytically for the case of a mixture of many beta energies - which is the case at hand. However, since $D\infty$ for each beta energy is known (from the calculation of the semi-infinite source), $D_{E(v)}$ for each beta energy at a given volume may be determined. All contributions to the total dose at a given volume are then added together.

The volumes evaluated in this analysis were 10^3, 10^4, 10^5, and 10^6 cm^3. Table J.E-1 summarizes the semi-infinite volume for each beta energy group. Table J.E-1 also indicates the beta dose reduction factor for each of the beta energy groups at the finite beta volumes of interest. A plot of the integrated 6-month doses for these finite beta volumes is shown in Figure J.E-1. These results reflect the reduction in beta air dose from the semi-infinite medium air dose to a finite volume medium air dose. The integrated beta infinite airborne dose for the reactor building as a function of time post-loss-of-coolant accident (LOCA) is shown in Figure J.E-2.

Table J.E-1

Dose Rate Reduction Factors for the Post-Loss-of-Coolant Accident Beta Energy Groups at Finite Volumes

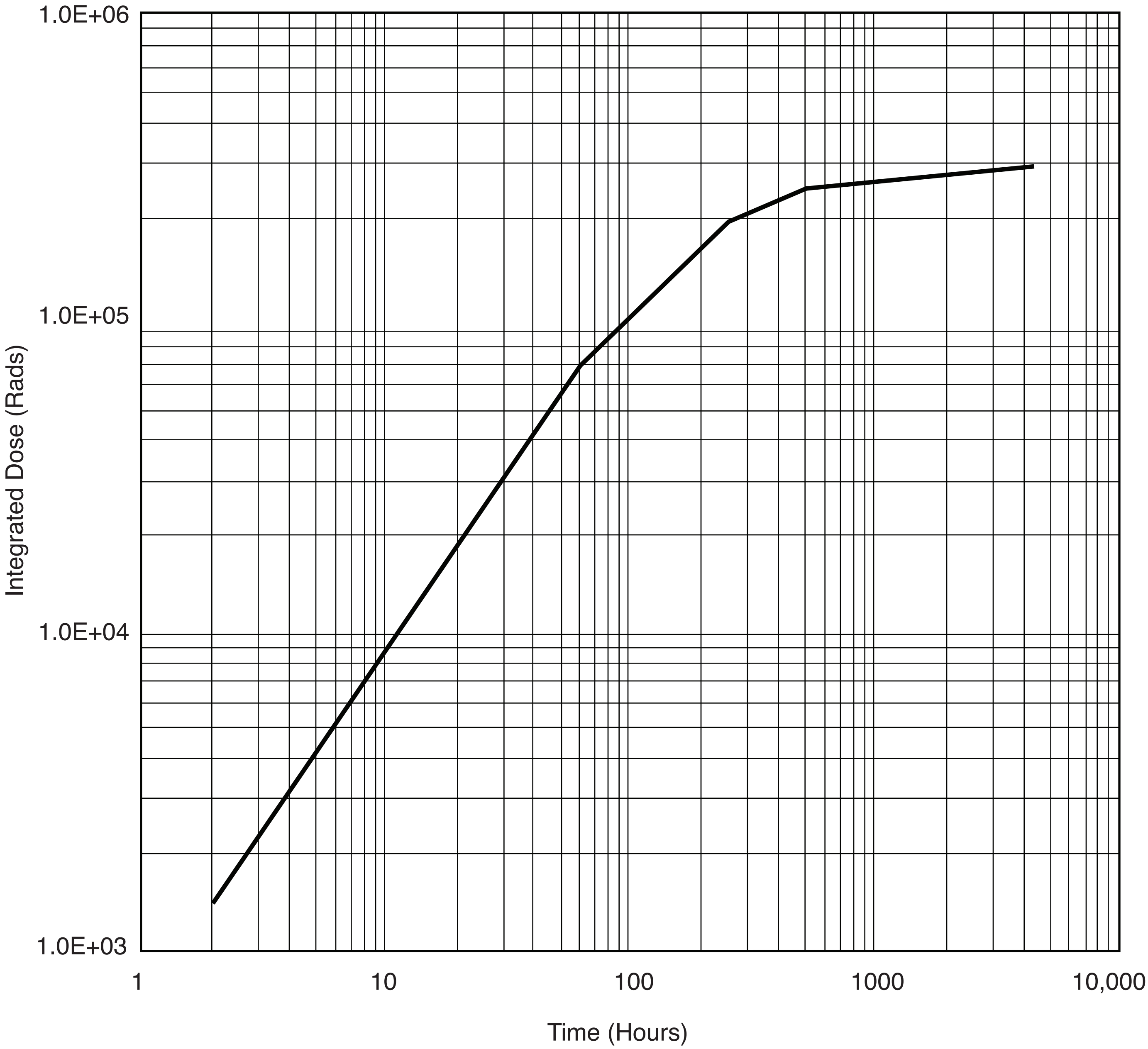
Energy Group (MeV)	V_E (cm^3)	$\frac{D(V)}{D_\infty}$ for Volumes			
		10^3 cm^3	10^4 cm^3	10^5 cm^3	10^6 cm^3
0.02 - 0.10	120.0	1.0	1.0	1.0	1.0
0.10 - 0.20	4.08×10^5	0.486	0.763	0.960	1.0
0.20 - 0.40	8.58×10^6	0.260	0.478	0.755	0.955
0.40 - 0.70	1.36×10^8	0.127	0.254	0.468	0.744
0.70 - 1.0	1.04×10^9	0.0695	0.144	0.284	0.513
1.0 - 1.3	3.46×10^9	0.0467	0.0979	0.199	0.380
1.3 - 1.6	8.18×10^9	0.0348	0.0735	0.152	0.299
1.6 - 2.0	1.59×10^{10}	0.0276	0.0585	0.122	0.244
2.0 - 2.5	3.20×10^{10}	0.0215	0.0457	0.0960	0.195
2.5 - 3.0	6.47×10^{10}	0.0167	0.0356	0.0752	0.155



Columbia Generating Station
Final Safety Analysis Report

Total Integrated Beta Cloud Airborne Dose
as a Function of Size

Draw. No. 960222.64	Rev.	Figure J.E-1



Integrated Beta Infinite Airborne Dose for the Reactor Building

Draw. No. 960222.65 | Rev. | Figure J.E-2

Attachment J.F

PRIMARY CONTAINMENT ANALYSES

J.F.1 STATEMENT OF PROBLEM

It is required by NRC regulations (NUREG-0737 and NUREG-0588, References J.7-5 and J.7-2) that safety-related equipment be qualified to withstand the radiation environment in which they are located for the 40 years of normal plant operation plus for the 6 months following a postulated design basis loss-of-coolant accident (LOCA). This attachment presents a summary of the evaluation of the radiation environment inside the primary containment of Columbia Generating Station (CGS) during normal plant operation and for the 6 months following the postulated LOCA. This attachment also calculates the maximum integrated dose due to those radiation sources.

J.F.2 BASIC APPROACH

NUREG-0737 offers two approaches for evaluating the qualification of equipment within primary containment; pressurized versus depressurized reactor coolant system, with the more conservative to be considered the base case. Both cases assume the same source (100% noble gas, 50% halogens, and 1% particulates of the core inventory). The difference between the two is that in the pressurized case, the source is assumed to remain in the reactor coolant system for the first 17 hr (Reference J.7-44) after the accident and then is assumed to be released into the primary containment. In the depressurized case, there is assumed to be an instantaneous release of 100% of the core noble gases and 50% of the core halogens to the free volume of the primary containment (Reference J.7-45). It is also assumed that 50% of the core halogens and 1% of the core solids are released to the reactor coolant and the suppression pool. This causes some double counting of halogens and hence some conservatism, since only 50% of the core halogens need ever be considered for release after a LOCA.

Both scenarios, the pressurized and depressurized were evaluated and it was determined that for CGS the depressurized case results in higher integrated doses (References J.7-46, J.7-50, and J.7-54). Therefore, it was considered to be the base case.

Due to the large number of C1E* components inside primary containment, it was deemed impractical (from both scheduling and cost considerations) to calculate the integrated dose to each piece of equipment. Therefore, it was decided to calculate the worst point dose from each of the major sources in the drywell and wetwell, and then to sum these for a conservative estimate of the total integrated dose. This methodology for determining a worst-case dose for equipment in the drywell is not valid for the region inside the sacrificial shield wall or under

* Environmental qualification (EQ) of safety-related mechanical equipment has been eliminated from the overall CGS EQ program (SRM).

the reactor pressure vessel. A point-specific radiation dose calculation is required for all components present in either of these two regions.

J.F.3 DRYWELL

The integrated dose from each of the major sources to the drywell is tabulated in Table J.F-1. All values are the maximum dose for each source considered. Since the maximum dose does not occur at the same location or the same time from all sources, it is not appropriate to sum them to obtain the total integrated dose. All of the maximum doses calculated cannot be present for a particular accident. The highest dose (7.4×10^7) is calculated for a depressurized reactor coolant system.

This dose is conservative since all of the source contributors summed do not have the maximum dose at the same location. If it were determined that certain pieces of equipment could not withstand the maximum dose, a more detailed calculation would unquestionably result in an integrated dose of lower than 7.4×10^7 rads. A lower bound for the more detailed calculation would be about 10^7 rads.

One major factor regarding the airborne contribution needs to be addressed here to understand the results in Tables J.F-1 and J.F-2. Of the total airborne contribution (3.5×10^7 rads) slightly over 50% of it is due to photons which have an energy of less than 0.045 MeV. These photons are readily attenuated. As such, virtually any amount of shielding will result in a reduction by a factor of approximately two in the total airborne dose. Such an example is the smallest size conduit used in containment which has a wall thickness of 0.179 in.

This is not the only conservatism in the calculation; however, it is the most noteworthy. The following section addresses the individual contributors, assumptions, sources, models, etc., used to calculate the integrated dose.

J.F.3.1 Sources

There are six major radiation sources to the equipment in the drywell. Two of these sources are present during normal operation and four sources are present after a LOCA. They are

	Normal Ops	Sources
Rx Core	Normal Ops	Neutrons emanating directly from the reactor core and the resultant capture gammas.

Systems	Normal Ops	The following systems are the main sources of radiation during normal plant operation: a. Residual heat removal (RHR) system, b. Reactor water cleanup (RWCU) system, c. Main steam (MS) system, and d. Reactor recirculation (RRC) system.
Systems	Post-LOCA	In addition to the systems considered under normal operation, (except for the MS) the following systems were also considered post-LOCA: a. High-pressure core spray (HPCS), b. Low-pressure core spray (LPCS), and c. Reactor core isolation cooling (RCIC).
Airborne	Post-LOCA	Airborne radiation from radionuclides (noble gases and halogens) which are postulated to be released into the primary containment atmosphere following a LOCA.
Plateout	Post-LOCA	Plateout on surfaces within containment. This consists of radioactive iodines which are initially airborne and subsequently plateout (Reference J.7-34).
Wetwell	Post-LOCA	The radionuclides contained within the wetwell as a result of the blowdown after the accident.

J.F.3.1.1 Reactor (Normal Operation - Drywell)

There exists a general radiation field inside primary containment due to normal plant operation. Part of this field is due to neutron leakage from the reactor core. A fraction of those neutrons penetrate the reactor vessel into the reactor cavity. Some will traverse vertically while others will penetrate the sacrificial shield wall. In addition, secondary gammas will be generated from neutron interaction with materials along their path.

ANISN, a one-dimensional discrete ordinates computer code was used to calculate the transport of these neutrons, and the generation of secondary gamma rays (Reference J.7-55).

The total neutron and gamma dose rates outside the sacrificial shield wall at core mid-plane are calculated to be

a. 5.7 rad/hr neutron, and
b. 50 rad/hr gamma

An estimate was made to determine the axial variation of the dose rate based on geometric and material attenuation factors. The approximate dose rate reduction factors are shown in Table J.F-3 as a function of distance from the core mid-plane.

J.F.3.1.2 Systems (Normal Operation - Drywell)

During normal operation, a radiation field exists within containment due in part to radioactivity contained within the piping inside primary containment.

The single major source within the piping is ^{16}N [produced by the (n,p) ^{16}O ^{16}N reaction within the core]. The dose from other sources such as fission products, corrosion products, etc., are too small compared to ^{16}N to be considered.

Calculations were done to determine the dose rate to which equipment was exposed. The results indicated that the dose rate ranged from a high of 35 rad/hr to a low of 0.36 rad/hr. These calculations were performed with KAP-V and QAD-BR. They took into account the following systems: RHR, RWCU, MS, and RRC (References J.7-47, J.7-48, and J.7-49). The ^{16}N source used was 40 µCi/g (FSAR Table 11.1-4) maximum. This is the source strength of the ^{16}N in the coolant exiting the reactor. Based on this initial source, the source strength for the pipes of the systems considered was evaluated, and the dose calculations were then performed.

J.F.3.1.3 System (Post-Loss-of-Coolant Accident) - Drywell

The dose rate calculations for systems post-LOCA were performed using a method similar to that used for the systems under normal operation with two exceptions. The first was that in addition to the RHR, RWCU, and RRC systems, the HPCS, LPCS, and RCIC systems were also included. The second exception was that a different source was used (References J.7-48 and J.7-49). After a LOCA, the predominant source past the first minute or so is the assumed fission product release from the core. The ^{16}N inventory, with a 7.1-sec half-life decays away in less than a minute once the (n,p) ^{16}O ^{16}N reaction stops occurring (after the reactor shuts down).

For the base case, i.e., the depressurized case, it was assumed that 50% of the core halogens and 1% of the core solids were released and distributed within the suppression pool and the reactor coolant systems (References J.7-51, J.7-52, and J.7-53). As noble gases were produced by the radioactive decay of the halogens, they were discounted on the premise that they would be released from the liquid to the gas rapidly. The released inventory is then decayed for 37 discrete time intervals out to 6 months (these are given in Table J.F-7). An average source strength is then calculated for the 6-month period. The source strength is given in Table J.F-4.

J.F.3.1.4 Airborne - Drywell

A nonmechanistic accident scenario was postulated in calculating the airborne source. It was assumed that after 1000 days of operation at 3556 MWt (105% of core power), 100% of the noble gases and 50% of the halogens contained within the reactor core are instantaneously released. After the release, no additional contribution of either noble gases or halogens is considered. Also, plateout of halogens is considered (see Section J.F.3.1.5). The average airborne source strength is given in Table J.F-5.

The above source is calculated via the ORIGEN2 computer code. After the source strength was determined, the dose rate was calculated using the QAD-CG computer code. Details of the model and the calculation are discussed in Section J.F.5. The value for the airborne contribution presented in Table J.F-1 represents the dose rate at a point within the drywell which is predominantly surrounded by air. This point was chosen because of the absence of structural steel, piping, etc., surrounding the dose point. This would result in an upper limit dose rate which could be expected to occur in the drywell.

The effect of the shielding afforded by the structural steel, piping, etc. (i.e., "shadow shielding"), within containment was considered. Advantage was taken of "shadow shielding" when considering the contribution of the more distant airborne sources (References J.7-48 and J.7-49). This significantly reduces the dose rate compared to the case where "shadow shielding" is not employed. See Section J.F.5 for modeling of "shadow shielding."

J.F.3.1.5 Plateout - Drywell

The basis for determining the plateout source is 50% iodine inventory released after 1000 days irradiation at a power level of 3556 MWt. However, the plateout source is only those iodines which are removed from the airborne source and assumed to plateout on the surfaces within containment. As such, plateout removes sources from the airborne source, and this was accounted for in the calculations. It was assumed, however, that the noble gases generated by the decay of the plated out halogens (I $\rightarrow$ Xe and BR $\rightarrow$ KR) are instantaneously released and are mixed within the free volume of the drywell. In this manner, both the airborne and plateout sources are determined with no "double counting" of nuclides. The plateout source is given in Table J.F-6.

When the halogens are initially released, not all of them are considered available to plateout. Of the halogens released, 2.0% are in the form of organic compounds, and 2.5% are in the form of particulates (Reference J.7-2); and both of these forms are assumed not to plateout.

The remaining 95.5% are considered to be in an elemental state of which one-two hundredth remain airborne and the rest plateout. Therefore, no more than 95% of the released halogens can ever plateout. The plateout was assumed to occur with an effective deposition velocity of 0.05 cm/sec. This translated into an effective half-life of 1.01 hr^{-1} (References J.7-34 and J.7-45). Given this half-life, the limit of a reduction of a factor of 200 is attained in slightly over 5 hr. After that time, the percentage of plated out halogens remains constant at 95%.

The dose calculations were performed with the computer code QAD-CG, incorporating a model similar to that used for the airborne dose. See Section J.F.5 for discussion of model and calculations.

Initial calculations were performed with the total plateout being distributed over: (1) the drywell lateral surface, top, and bottom; (2) inner, outer and top surfaces of the sacrificial shield wall; and (3) heat reflector of pressure vessel surface. Given this distribution area, the maximum dose rate calculated was 7.04×10^3 rad/hr. However, when the remaining surface areas within containment (i.e., equipment, piping, structural steel, etc.) were considered, the area over which the source would be plated out increased sevenfold. A counter-balancing effect to this reduction in plated out concentration was that the source would be more universally distributed around any given receiver. It was estimated that the net effect would reduce the calculated maximum dose rate by a factor of approximately three.

It is noted that the energy spectrum for the plateout source is significantly harder than that of the airborne. As such, the comments in Section J.F.3 regarding low energy photons are not completely applicable.

J.F.3.1.6 Wetwell - Drywell

The wetwell was also considered as a source to the drywell. However, due to distance, self-attenuation, and the available shielding from the 2-ft-thick diaphragm floor, its contribution to the drywell was negligible.

It was assumed that 50% of the halogens and 1% of the particulates from the core were entrained in the water in the suppression pool. This is the same source used for the systems post-LOCA. The air space volume above the suppression pool was assumed to have the same volumetric source strength as the drywell air space. These are conservative premises since only a total of 50% of the core halogens are assumed to be released after the accident.

J.F.4 WETWELL

The results for the wetwell are given in Table J.F-2. Doses were calculated for detector points both within the suppression pool as well as in the free volume above it using QAD-CG, applying the same modeling techniques as was used in the drywell.

With regard to the airborne contribution, the volumetric source strength is the same as the drywell airborne source and the comments in Section J.F.3.1 regarding the low-energy photons applicability to the wetwell.

There does exist some double counting of nuclides in the wetwell analysis. The airborne source is 100% noble gases and 50% halogens, released into the containment (wetwell and drywell) free volume. For the suppression pools the source is 50% halogens and 1% particulates. Since only 50% of the total core halogens are assumed to be released after an accident, they are double counted (the effect is small, however, because of the shielding offered by the suppression pool water.) Another conservatism in the airborne source in the wetwell is that, since the path for the wetwell airborne sources is via the downcomers and then up through the suppression pool, some halogens are expected to be entrained in the water during this transfer (this was not considered in the calculation.) The result would have been a smaller airborne source and in turn a smaller dose.

J.F.4.1 Sources

There are three sources of radiation to the equipment in the wetwell, all of which are present only after an accident.

a. Airborne

   The airborne source is present as a result of the initial blowdown into the suppression pool via the downcomers,

b. Plateout

   Plateout of halogens onto the surfaces in the wetwell (i.e., containment, downcomers, etc.), and

c. Suppression Pool

   The radionuclides contained within the suppression pool as a result of the blowdown after the accident.

J.F.4.1.1 Airborne - Wetwell

The airborne source, on a specific volume basis, is equal to the airborne source in the drywell (i.e., 100% noble gas and 50% halogen released into the total primary containment immediately following a LOCA). However, the amount of "shadow shielding" within the wetwell is much less than in the drywell. Hence, the contribution from sources further away is greater. This factor accounts for the increased dose rate in the wetwell with respect to the drywell (due to airborne sources). Dose calculations in the wetwell were done in similar manner as for the drywell (i.e., using QAD-CG).

J.F.4.1.2 Plateout - Wetwell

As in the drywell, the source of the plateout in the wetwell is the halogens. However, the area available for plateout is smaller in the wetwell than the drywell. This results in a dose rate in the wetwell slightly more than double that in the drywell.

J.F.4.1.3 Suppression Pool - Wetwell

The source in the suppression pool was assumed to be 50% of the halogens and 1% of the particulates instantaneously released from the core into the pool and the reactor coolant system. It is further assumed that as noble gases are produced by the decay of the halogens (I→Xe and Bγ→Kγ), they "bubble out" of the pool, hence they are not considered a source term. Dose rates both in the suppression pool as well as in the wetwell free volume were calculated using QAD-CG.

J.F.5 QAD-CG MODEL

The QAD-CG computer program was used to calculate dose rates for both the airborne source as well as the plateout. In both cases, i.e., airborne and plateout, similar modeling techniques were used. This section defines the modeling used in both calculations (with only the drywell used for illustrative purposes).

The QAD-CG computer code makes use of a geometry package, which allows the user to model a calculation with the use of predetermined geometric bodies. The user defines a set of geometric "bodies" (boxes, truncated cones, spheres, cylinders, etc.) and using these "bodies," the user defines "zones" by intersection or forming unions of them to build the shapes desired in a manner analogous to "intersections" and "unions" when one deals with sets. The model is done three-dimensionally thereby allowing the user considerable flexibility. These "zones" are then what constitute the computer model. The parts of "bodies" that are not used have no effect on the model.

As an example, a dumbbell could be defined as the union of three "bodies": two spheres and a long, thin cylinder between them (see Figure J.F-1). Likewise, a hemisphere could be formed by intersection of a sphere with a box (see Figure J.F-1). In this manner, a complex model can be defined.

In our case, the basic model was defined as a truncated cone (approximating the containment shell) and two cylinders (approximating the sacrificial shield wall and the reactor vessel). Figure J.F-2 illustrates this in a sectional view. The free volume of the drywell was compartmentalized into cubes, 7 ft on a side. These cubes were formed by intersection of a series of tall rectangles, which are 7 ft on a side in cross-section, with cylinders at 7-ft high intervals. Each 7-ft high cylinder constitutes the elevational boundaries of what is referred to as a "layer" below. Combining these "bodies" appropriately one winds up with a truncated cone (containment) with two cylinders (i.e., sacrificial wall and reactor vessel) and the remainder of the volume forced with cubes (except on the boundary of the cone or cylinders). Figure J.F-3 illustrates this model, while Figure J.F-4 illustrates how the layer from el. 513 ft 6 in. to el. 520 ft 6 in. is modeled.

All major structures, pipes (6 in. and above), hangers, etc., within the drywell were then located, and the mass of steel in each cubicle determined. These were translated into average densities such that each cube had an average density assigned to it. These are illustrated on Figures J.F-5 to J.F-9 for the lower five layers; for the purpose of clarity, the densities shown are much cruder than the 41 used in the code. In those cubicles which are noted to have zero density, the density of air was assumed.

In Figures J.F-7 to J.F-9 a large void (air only) exists in the southwest (fourth) quadrant in layers 3 and 4. It was in this region that the airborne dose rate was calculated. This region provides us with a volume which is large enough so that the "shadow shielding" (smearing discrete shielding within a cubicle into an average density in the cubicle) beyond its boundary is justified.

Several runs were made using this model with various source volumes. Three runs were made placing the source terms within the elevational boundaries of layers 3, 4, and 5, respectively, and another run was made by placing the source from the lower elevational boundary of layer 1 up to the upper elevational boundary of layer 2. It was noted that >95% of the total dose contribution from these five layers came equally from layers 3 and 4. In other words, the further away the source layer, the smaller the contribution. Also, shadowing shielding in layers 1 and 2 provided sufficient attenuation as to make the contribution to the total dose negligible. The same is true also for all layers above layer 5.

Plateout was calculated in a similar manner, increasing the source until successive contributions became negligible. For the plateout, the dose point was taken near the sacrificial shield wall. Other points were also considered, but the dose rate near the sacrificial shield wall was found to be the maximum. Again, the dose point was taken between layers 3 and 4 to maximize the dose rate.

J.F.6 CODES

J.F.6.1 FSPROD

FSPROD is a computer program which calculates the inventory and activity of radioactive fission products, produced from the thermal fission of ^{235}U, as a function of fission rate and decay time after fission. The program is used in establishing the gross and specific gamma and beta activity of those fission products. The calculation incorporates 123 fission product nuclides and is based on Perkins and King data.

J.F.6.2 ORIGEN2

ORIGEN2 is a point depletion and decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide composition of materials contained therein. The code represents a revision and update of the original ORIGEN computer code. The general function of the ORIGEN2 computer code is to calculate the nuclides present in various nuclear materials by determining the buildup and depletion of nuclides during irradiation and decay. The code can also account for reprocessing (i.e., chemical separation) and continuous feed, removal, and accumulation of nuclear materials.

J.F.6.3 QAD-BR

QAD-BR is a point kernal computer code designed to evaluate gamma penetration of various shield configurations. It is a modification of QAD-P5A; i.e., it has no capability for neutron calculations. The program provides an estimate of the uncollided and collided gamma flux, dose rate, energy deposition, and other quantities which result from a point-by-point representation of volume-distribution source of radiation.

J.F.6.4 QAD-CG

QAD-CG is also a modification of the QAD-P5A computer program. It is similar to QAD-BR in application with the major difference being in the geometry description. QAD-CG makes use of a combinatorial geometry package originally developed for MORSE. It is one of the more versatile geometry packages to be available in the QAD family of computer codes.

J.F.6.5 KAP-V

KAP-V is a hybrid of the QAD computer code. Analytically, it is identical to QAD as it is a point kernal code. The major differences are changes in input allowing more flexibility in running successive cases. It also has internal libraries for attenuation and buildup data which can be used by default for convenience.

J.F.6.6 ANISN

ANISN is a one-dimensional Sn transport code with anisotropic scattering. It allows for the solution of the transport equation for neutrons and photons using the discrete ordinate method.

Table J.F-1[a]

Integrated Dose in Drywell

Source	Maximum Average Dose Rate[b] (rad/hr)	Exposure Time	Dose[b] (rad)
Reactor	5.6×10^{1}	32 years[c]	1.6×10^{7}
Systems - normal	3.5×10^{1}	32 years[c]	9.9×10^{6}
Systems - LOCA	---	6 months	3.2×10^{6}
Airborne	---	6 months	3.7×10^{7}
Plateout	---	6 months	1.0×10^{7}
Suppression pool	---	6 months	$< 4.5 \times 10^{4}$

[a] Not valid for regions inside the sacrificial shield wall or under the reactor pressure vessel (a point specific radiation calculation is required for components in these two regions).

[b] Maximum dose rate from individual contributors does not necessarily occur at the same location or for the same accident.

[c] (40-year plant life) x (0.8) availability to account for down time.

Table J.F-2

Integrated Dose in Wetwell

Source	Maximum Average Dose Rate[a] (rad/hr)	Exposure Time	Dose[a] (rad)
Dose above suppression pool			
Airborne	1.8×10^4	6 months	8.2×10^7
Suppression pool	2.0×10^2	6 months	9.1×10^5
Plateout	2.7×10^3	6 months	1.2×10^7
Dose within suppression pool			
Airborne	2.9×10^2	6 months	1.4×10^6
Suppression pool	5.5×10^2	6 months	2.5×10^6

[a] Maximum dose rate from individual contributors does not necessarily occur at the same location.

Table J.F-3

Approximate Dose Rate Reduction Factor
Versus Distance from Core Mid-Plane
for Reactor Integrated Dose

Distance (ft)	Reduction Factor
0	1.0
5	0.5
10	0.02
15	1×10^{-5}

Table J.F-4

Suppression Pool and System
(Loss-of-Coolant Accident) Liquid Source Terms
0-6 Month Average After Loss-of-Coolant Accident

MeV	MeV/sec	MeV/cm^3-sec[a]
0.015	1.8E+14	4.4E+4
0.025	3.5E+14	8.5E+4
0.0375	4.8E+14	1.2E+5
0.0575	1.4E+14	3.5E+4
0.085	5.8E+14	1.4E+5
0.125	2.2E+15	5.3E+5
0.225	4.0E+15	9.6E+5
0.375	4.8E+16	1.2E+7
0.575	5.5E+16	1.3E+7
0.85	7.2E+16	1.7E+7
1.25	1.5E+16	3.7E+6
1.75	1.8E+16	4.3E+6
2.25	2.1E+15	5.1E+5
2.75	9.1E+14	2.2E+5
3.5	1.1E+14	2.6E+4
5.0	7.4E+13	1.8E+4

[a] Volume considered was that of the suppression pool plus that of the reactor coolant system.

Table J.F-5

Airborne Source Terms
0-6 Month Average After Loss-Of-Coolant Accident

MeV	MeV/sec	MeV/cm^3-sec[a]
0.015	2.5E+14	2.5E+4
0.025	2.1E+14	2.1E+4
0.0375	6.9E+15	7.1E+5
0.0575	3.0E+13	3.0E+3
0.085	1.4E+16	1.4E+6
0.125	3.7E+13	3.8E+3
0.225	4.4E+15	4.5E+5
0.375	3.0E+15	3.1E+5
0.575	4.0E+15	4.1E+5
0.85	3.2E+15	3.2E+5
1.25	3.4E+15	3.5E+5
1.75	2.6E+15	2.7E+5
2.25	3.9E+15	4.0E+5
2.75	6.7E+14	6.9E+4
3.5	2.1E+14	2.2E+4
5.0	9.5E+13	9.7E+3

[a] Volume considered was total; i.e., drywell plus wetwell free volume.

Table J.F-6

Drywell Plateout Source Terms
0-6 Month Average After Loss-Of-Coolant Accident

MeV	MeV/sec	MeV/cm^3-sec[a]
0.015	3.6E+13	6.3E+5
0.025	1.8E+14	3.2E+6
0.0375	5.4E+13	9.5E+5
0.0575	2.0E+13	3.6E+5
0.085	3.2E+14	5.7E+6
0.125	1.9E+13	3.4E+5
0.225	2.8E+15	4.9E+7
0.375	4.4E+16	7.7E+8
0.575	2.4E+16	4.2E+8
0.85	7.6E+15	1.3E+8
1.25	9.6E+15	1.7E+8
1.75	3.4E+15	5.9E+7
2.25	5.2E+14	9.2E+6
2.75	7.3E+12	1.3E+5
3.5	1.3E+13	2.3E+5
5.0	2.9E+11	5.1E+3

[a] These values should be reduced by a factor of seven when all structural, component and equipment surfaces in containment are considered.

Table J.F-7

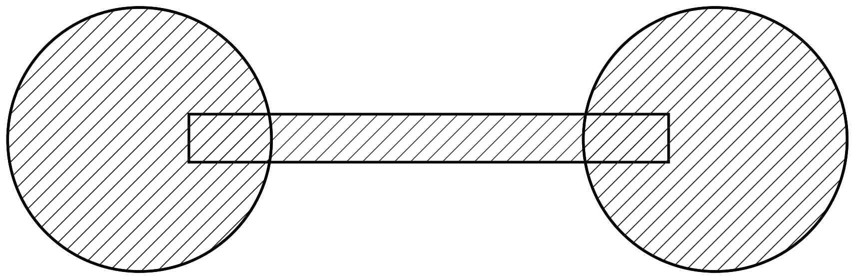
Time Mesh Spacing Used in Source Calculations
(Minutes)

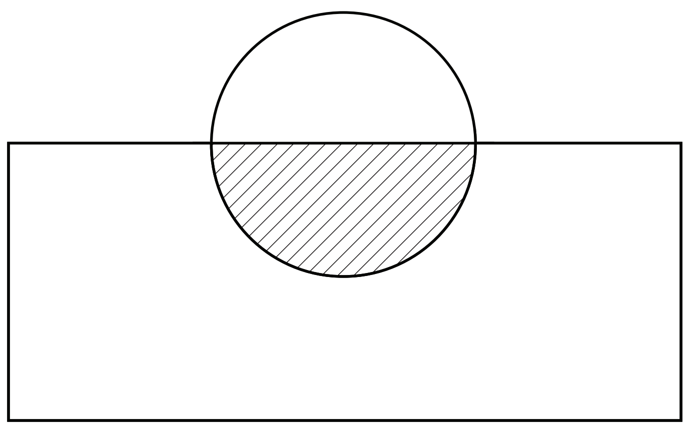
0	640	28800
20	800	36000
40	960	43200
60	1120	57600
80	1280	72000
100	1440	86400
120	2160	108000
180	2880	129600
240	3600	151200
300	4320	172800
360	5040	216000
420	5740	259200
480	14400	

Table J.F-8

Source Energy Group Structure

Lower Boundary (MeV)	Upper Boundary (MeV)	Average Energy (MeV)
0.00	0.02	0.015
0.02	0.03	0.025
0.03	0.045	0.0375
0.045	0.07	0.0575
0.07	0.10	0.085
0.10	0.15	0.125
0.15	0.30	0.225
0.30	0.45	0.375
0.45	0.70	0.575
0.70	1.0	0.85
1.0	1.5	1.25
1.5	2.0	1.75
2.0	2.5	2.25
2.5	3.0	2.75
3.0	4.0	3.5
4.0	6.0	5.0





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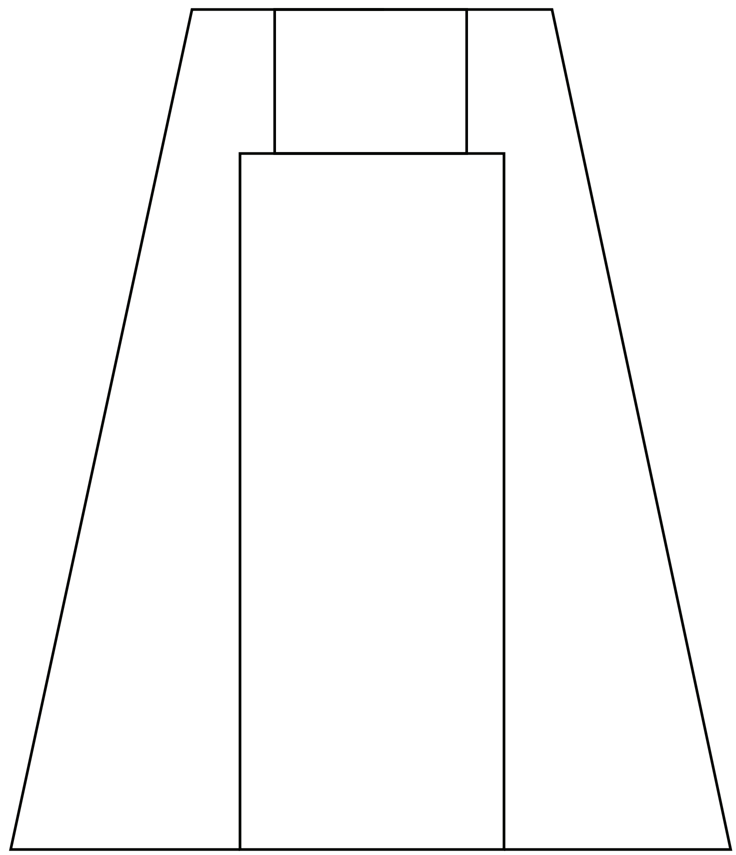
Geometry Examples

Draw. No. 970187.69 | Rev. | Figure J.F-1



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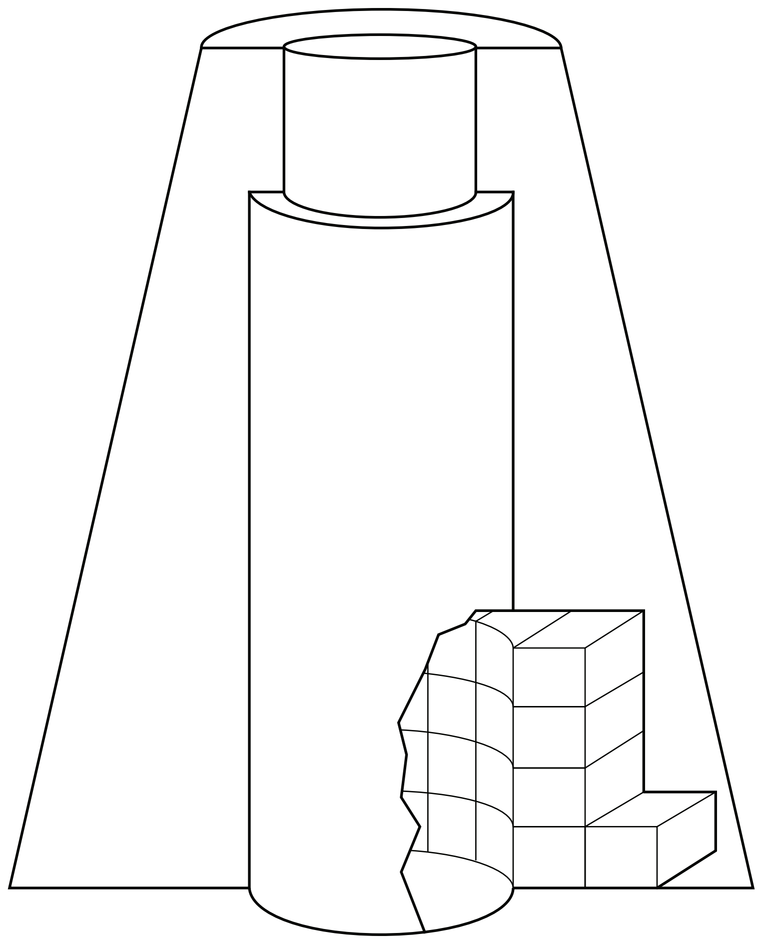
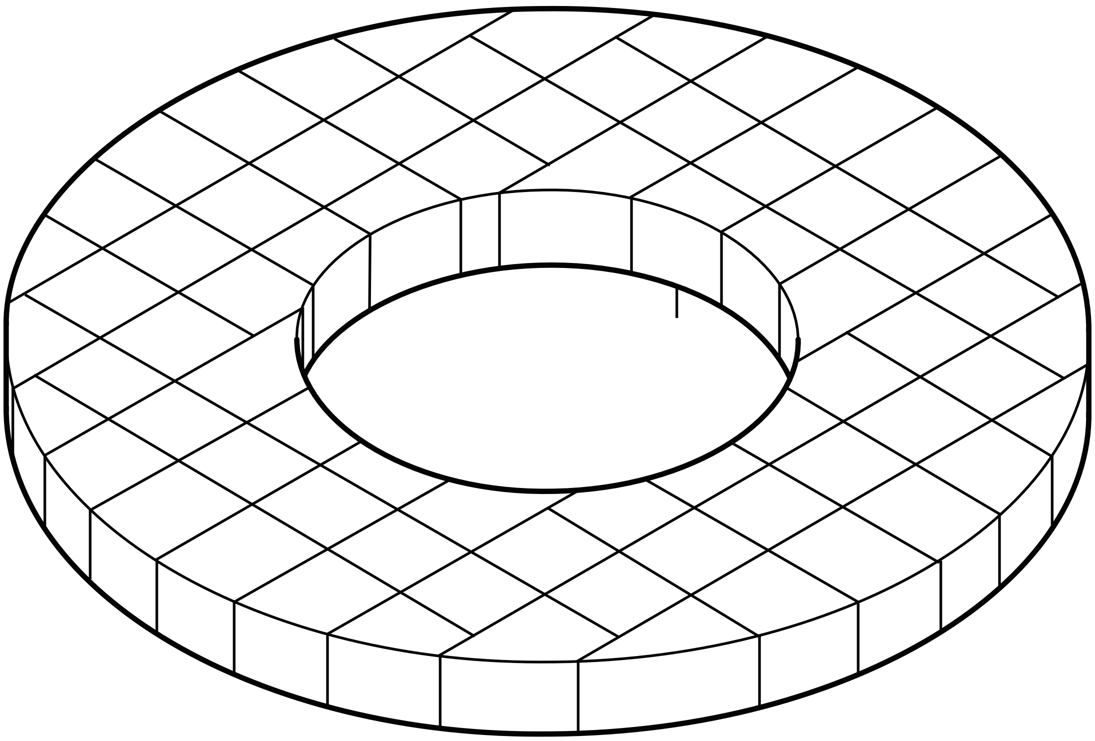
Basic QAD-CG Drywell Model

Draw. No. 970187.70 | Rev. | Figure J.F-2



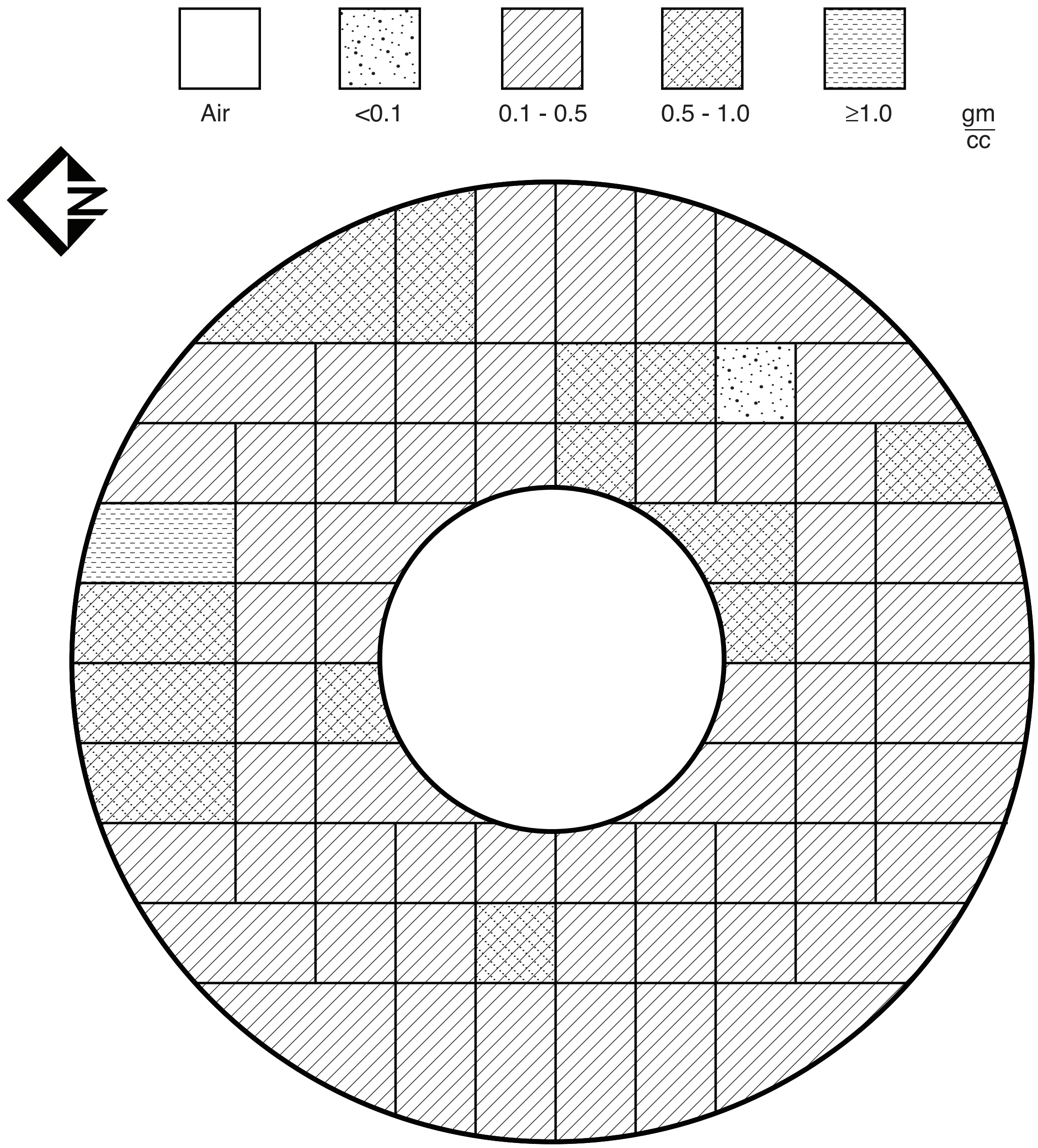
Isometric of Drywell Model

Draw. No. 970187.71 | Rev. | Figure J.F-3



Isometric of El. 513 ft 6 in. to 520 ft 6 in.

Draw. No. 970187.72 | Rev. | Figure J.F-4



Illustrative Only - Code Used 41 Density Groups

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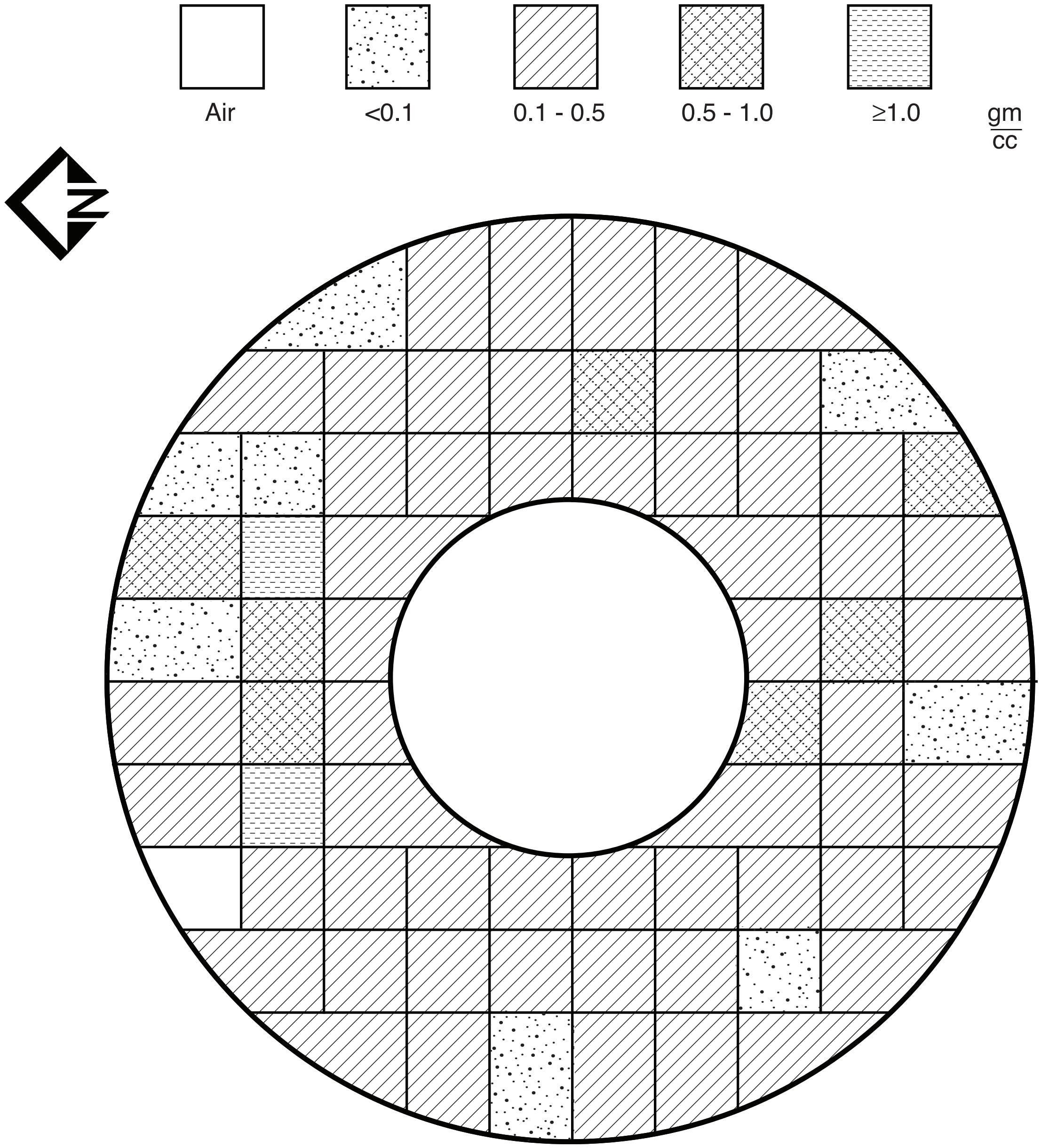
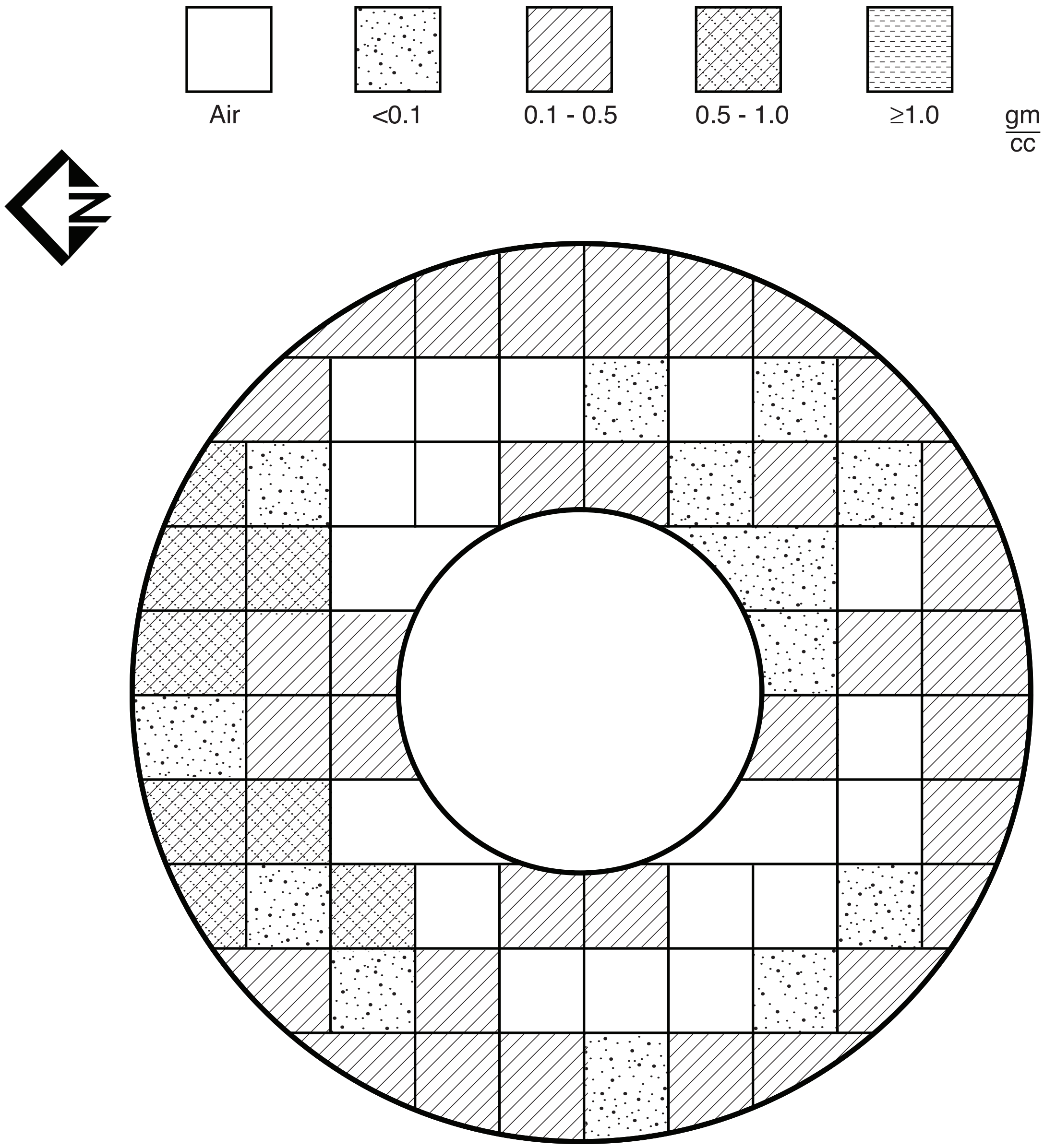
Plan at El. 499 ft 6 in.

Draw. No. 970187.73 | Rev. | Figure J.F-5



Illustrative Only - Code Used 41 Density Groups



Illustrative Only - Code Used 41 Density Groups

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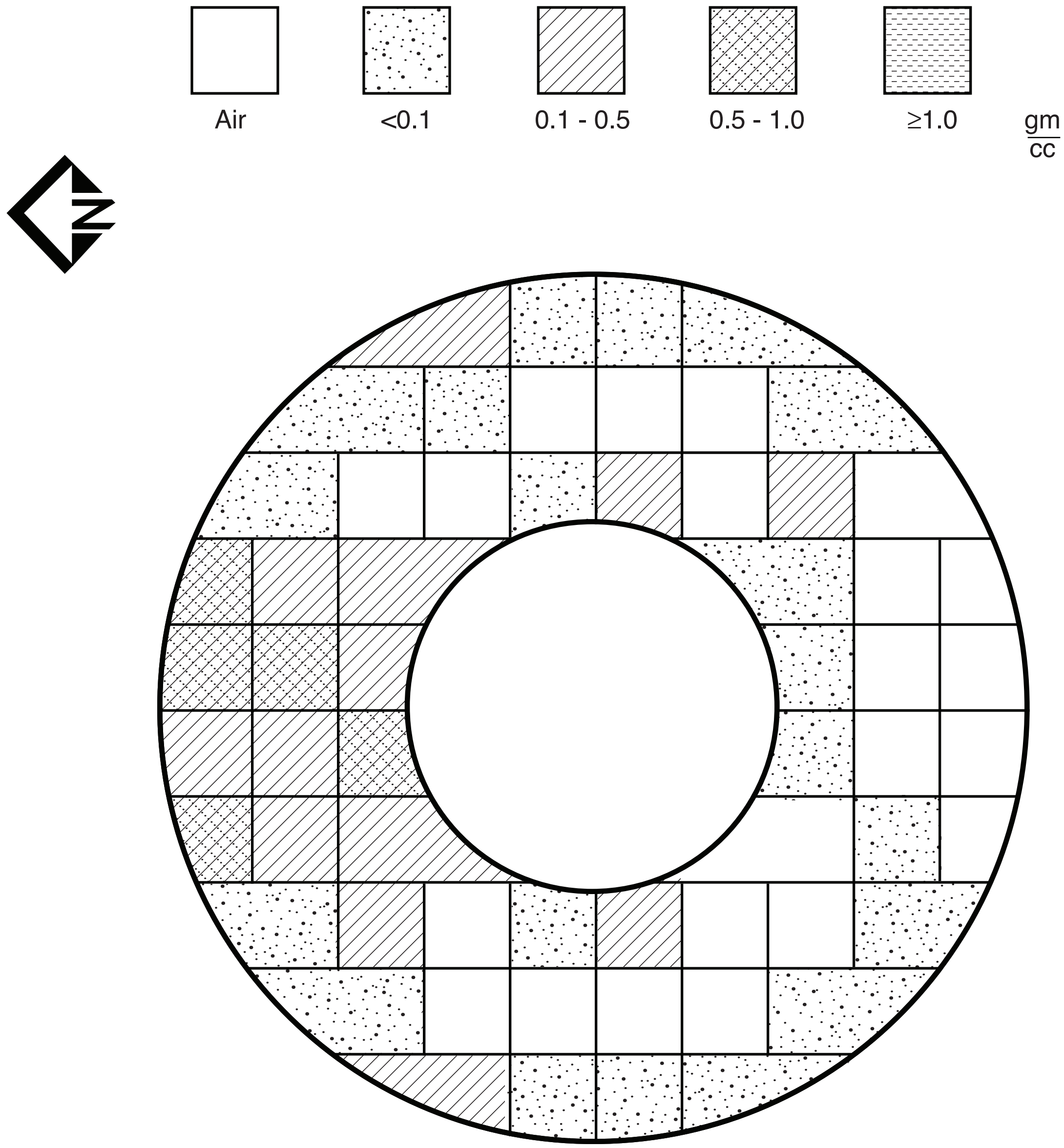
Plan at El. 513 ft 6 in.

Draw. No. 970187.75 | Rev. | Figure J.F-7



Illustrative Only - Code Used 41 Density Groups

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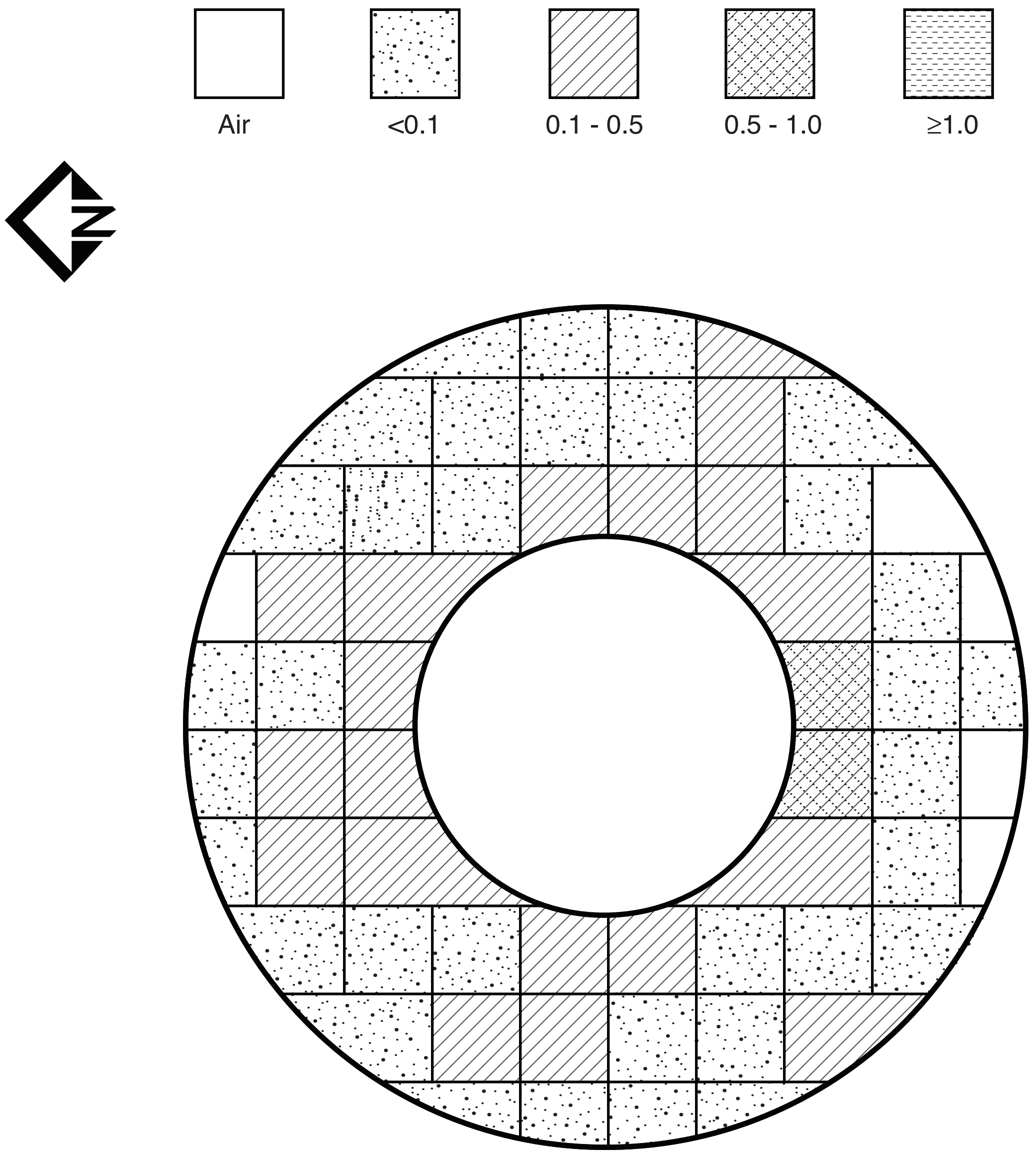
Plan at El. 520 ft 6 in.

Draw. No. 970187.76	Rev.	Figure J.F-8



Illustrative Only - Code Used 41 Density Groups

Plan at El. 527 ft 6 in.

Attachment J.G

BETA DOSE CONTRIBUTION IN PRIMARY CONTAINMENT

The source volume used for the beta dose analysis in primary containment is a sphere surrounded by a shell of sufficient thickness to stop all outside beta particles from entering the source volume. This spherical source volume is conservative for any generalized source volume shape (the dose at the center of the sphere is higher than the dose at any point of any generalized source of equal total volume).

The assumptions used in the analysis are as follows:

a. Atmosphere inside the equipment casing is identical to the atmosphere in primary containment. This is conservative because there will actually be some delay in transport of the gaseous fission products into the equipment;

b. The initial beta source term used was 100% of core noble gases and 50% of core halogens (References J.7-2 and J.7-34);

c. Daughter products of the airborne noble gases and halogens are included in the calculation of the airborne dose which is conservative and was required by the use of ORIGEN2 as a source code (Reference J.7-8);

d. Plateout of halogens inside primary containment was utilized as allowed per Reference J.7-34. The dose contribution of fission products plated out on equipment casings was neglected. The deletion of dose contributions from fission products plated out on equipment casings is acceptable, since equipment surface areas are small relative to the available containment surface area. In addition, the betas emitted from plated out fission products would be absorbed in the equipment casing and, hence, would not affect internal components;

e. No primary to secondary containment leakage is assumed since it maximizes the beta source concentration in primary containment;

f. Activity is assumed to be uniformly distributed throughout the containment free volume which is reasonable, considering the mixing effects of the loss-of-coolant accident (LOCA) blowdown and the operation of the drywell fan coolers; and

g. A spherical volume representing the equipment casing will be used.

The beta dose to equipment is dependent on the internal volume size of the piece of equipment. The beta dose is determined through the use of an energy dependent geometry factor and a ratio of the internal equipment volume to an infinite cloud. The beta dose contribution is excluded from the worst case total integrated gamma doses of primary containment shown in Section J.6.1 and Tables J.F-1 and J.F-2. The beta dose contribution is also excluded from the value, pump, and fan tables for C1E/SRM equipment in a primary containment environment.

The discussion and development of beta dose rate variation due to beta energy distribution in a one-dimensional absorbing medium is also valid for primary containment analysis.

Thus, the dose as a function of volume radius is given by the dual relation:

$$D(r) = D\infty \frac{[1 - \exp(-\mu_E r)]}{[1 - \exp(-\mu_E r_E)]} \qquad 0 \le r \le r_E$$

This relation may be transformed to a function of volume by noting that $V = 4 \pi r^3/3$.

Since μ_E and r_E vary for each beta energy, this equation cannot be solved analytically from the case of a mixture of many beta energies, which is the case at hand. However, since $D\infty$ for each beta energy is known (from the calculation of the semi-infinite source), $D_{E(v)}$ for each beta energy at a given volume may be determined. All contributions to the total dose at a given volume are then added together.

The volumes evaluated in this analysis were 10^3, 10^4, 10^5, and 10^6 cm^3. Table J.G-1 summarizes the semi-infinite volume for each beta energy group. Table J.G-1 also indicates the beta dose reduction factor for each of the beta energy groups at the finite beta volumes of interest. A plot of the integrated post-LOCA doses for these finite beta volumes is shown in Figure J.G-1. These results reflect the reduction in beta air dose from the semi-infinite medium air dose to a finite volume air dose.

The integrated beta infinite airborne dose for the primary containment as a function of time post-LOCA is shown in Figure J.G-2.

The absorbed beta dose within a physical target is not always equal to the beta dose at a mathematical point in air at the surface of that piece of equipment. The beta ionization energy (dose) deposited on the surface of a solid object is distributed in a thin surface layer to a depth equal to the beta range in the material. The relative material penetration of the different beta energy groups is used to provide a total integrated LOCA dose as a function of material depth.

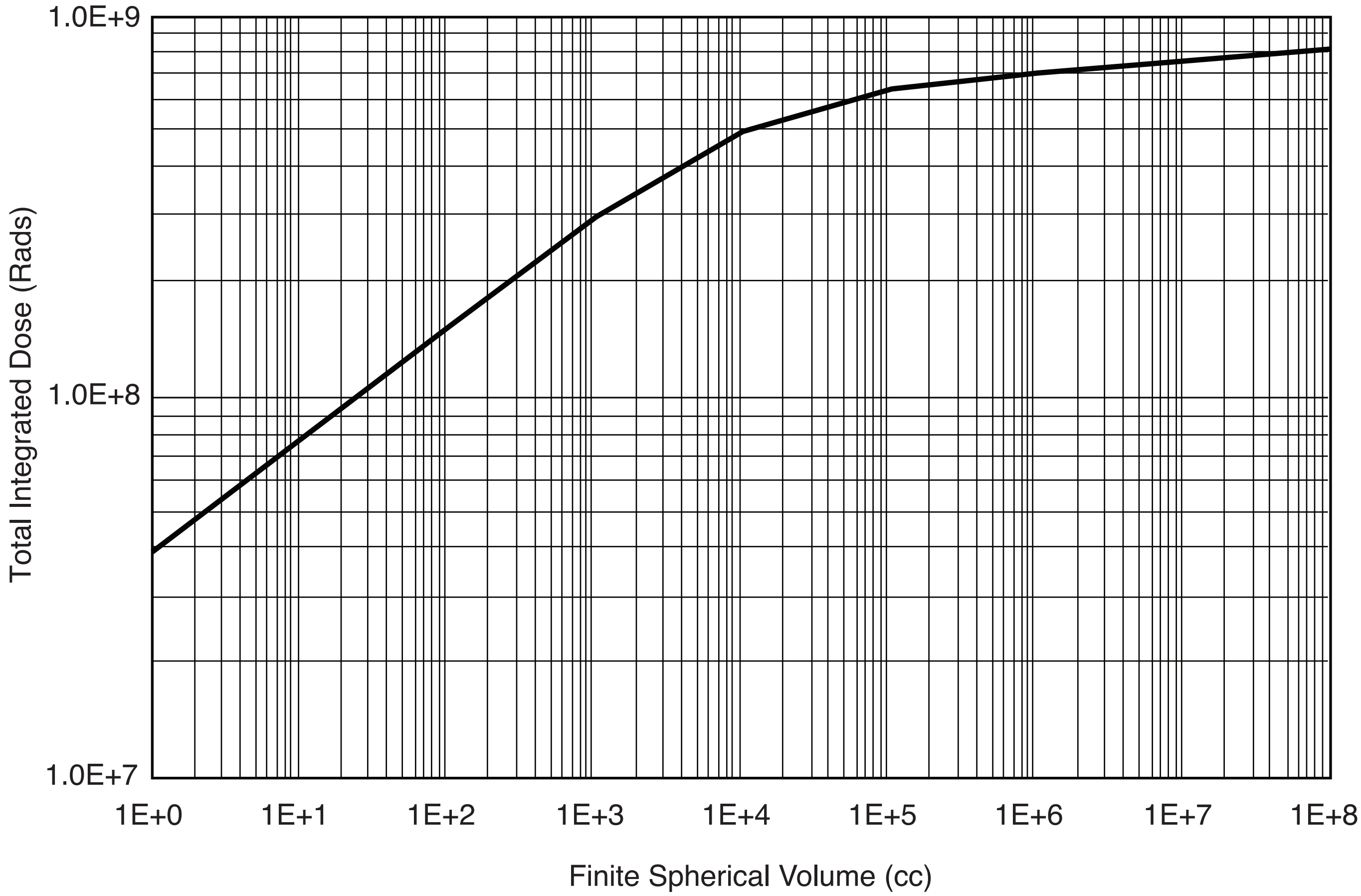
Finite volume beta dose reduction factors were determined for each of the 10 beta energy groups. These factors are used to provide total integrated LOCA dose as a function of material penetration to reduce volume exposure.

Thus, the integrated dose values (Figure J.G-1) can be used as the absorbed material dose with a standard order of magnitude for reduction for material beyond 0.030-in. thickness or a dose reduction versus thickness based on the range of beta penetration within the material can be calculated.

Table J.G-1

Dose Rate Reduction Factors for the
Post-Loss-of-Coolant Accident
Beta Energy Groups at Finite Volumes

		$\frac{D(V)}{D_\infty}$ for Volumes			
Energy Group (MeV)	V_E (cm^3)	10^3 cm^3	10^4 cm^3	10^5 cm^3	10^6 cm^3
0.02 - 0.10	120.0	1.0	1.0	1.0	1.0
0.10 - 0.20	4.08×10^5	0.486	0.763	0.960	1.0
0.20 - 0.40	8.58×10^6	0.260	0.478	0.755	0.955
0.40 - 0.70	1.36×10^8	0.127	0.254	0.468	0.744
0.70 - 1.0	1.04×10^9	0.0695	0.144	0.284	0.513
1.0 - 1.3	3.46×10^9	0.0467	0.0979	0.199	0.380
1.3 - 1.6	8.18×10^9	0.0348	0.0735	0.152	0.299
1.6 - 2.0	1.59×10^{10}	0.0276	0.0585	0.122	0.244
2.0 - 2.5	3.20×10^{10}	0.0215	0.0457	0.0960	0.195
2.5 - 3.0	6.47×10^{10}	0.0167	0.0356	0.0752	0.155



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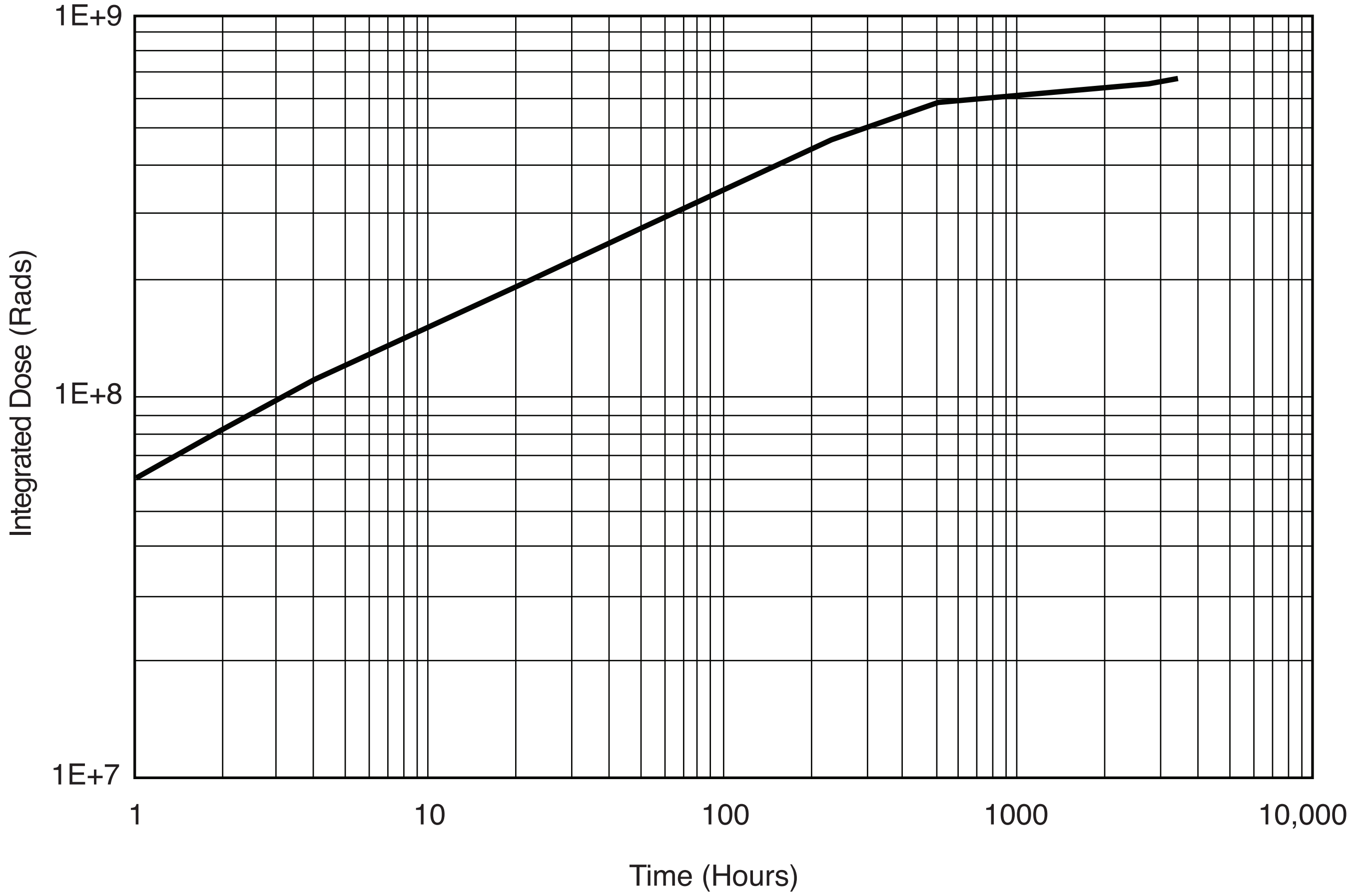
Total Integrated Beta Cloud Airborne Dose in Primary Containment as a Function of Size

Draw. No. 960222.58	Rev.	Figure J.G-1



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Integrated Beta Infinite Airborne Dose for Primary Containment

Draw. No. 960222.60 | Rev. | Figure J.G-2

Attachment J.H

VITAL AREAS AND ACCESS ROUTES ANALYZED FOR POST-LOSS-OF-COOLANT-ACCIDENT OPERATIONS

This attachment represents the methodology and assumptions used to determine the integrated dose to equipment and personnel for vital areas and access routes outside the reactor building during post-loss-of-coolant accident (LOCA) operations. The source term is the reactor building elevated vent with gaseous effluents being filtered by the standby gas treatment system (SGTS) prior to discharge to the atmosphere.

J.H.1 SOURCE OF RADIOACTIVITY TO THE REACTOR BUILDING ELEVATED VENT

Two contributions were considered as the source of the radioactivity to the reactor building elevated vent:

a. Leakage from the drywell to the reactor building and discharged via the SGTS to the reactor building elevated vent was assumed at a rate of

   0.5%/day = 2.1E-4/hr, and

b. Leakage from the assumed leaks on the main steam isolation valves (MSIVs) in the main steam tunnel was assumed at a rate of

   0.17%/day = 7.1E-5/hr (Reference J.7-56)

   Thus, the total leakage rate of activity from the primary system is assumed to be

   0.67%/day = 2.8E-4/hr.

J.H.1.1 Reactor Building Air Discharge Rate

All radioactivity considered outside the reactor building is assumed to discharge via the reactor building elevated vent.

The removal rate of the reactor building ventilation can be determined as follows:

$$\text{Removal rate} = \frac{\text{SGTS discharge rate}}{\text{Reactor building volume}}$$

SGTS discharge flow = 2430 ft^3/minute

Reactor building volume = $3.5E+6\ ft^3$

Thus, the removal rate is as follows representing one volume change per day:

$$\text{Removal rate} = \frac{(2430\ ft^3/min)(60\ min/hr)}{3.5E+6\ ft^3}$$

Removal rate = 4.2E-2/hr

This removal rate was used in the determination of radiation levels outside the reactor building.

J.H.2 POSTACCIDENT DESIGN DOSE (PADD)

A small computer program (PADD) was written to complete the calculations for the 18 nuclides over various time periods and sum the results. The equation used to determine the dose is as follows:

$$\text{Dose(rad)} = DF(j)\ (\frac{\chi}{Q_1} * TF * \frac{Q1_j + Q2_j}{3600}) \quad \text{(J.H-1)}$$

where

$Dose_{ji}$ = Rads from jth nuclide for the ith time period.

DF_j = Gamma dose factors for semi-infinite cloud $\frac{Rad * m^3}{Ci * hr}$ for jth nuclide.

χ/Q_1 = sec/m^3 for gaseous releases from the reactor building vent to the atmosphere for the ith time period.

RF = Removal fraction of activity via the standby gas treatment.

TF = 0.01 for particulates and iodines (99% efficiency or RF).

TF = 1.0 for noble gases (FSAR Section 6.5).

$Q1_j$ = Integrated activity of jth nuclide over ith time period that was released via leaks in the MSIVs (curies/hour).

$Q2_j$ = Integrated activity of jth nuclide over the ith time period that was released via leakage from the primary to secondary containment (curies/hour).

3600 = Conversion from hours to seconds.

J.H.2.1 Assumptions Used in χ/Q Calculation Methodology

The following equation from "Meteorology and Atomic Energy" (Reference J.7-31) was used to determine the χ/Q values shown in Table J.H-1.

$$\text{Dilution} = 2.22(M)\ \left(3.16 + 0.1\frac{S}{(Aex)^{1/2}}\right)^2 \frac{V_{mean}}{V_{ex}} \qquad \text{(J.H-2)}$$

$= F_B$ (building wake factor)

M = 1 if intake and exhaust same elevation

M = 2 if intake and exhaust separated by one floor

M = 4 if intake is in building wake cavity

S = shortest intake exhaust arc length

Aex = exhaust area

V_{mean} = mean approach flow

V_{ex} = mean exhaust flow

The intake was assumed to be for category F weather conditions with a V_{mean} = 1 meter/sec.

Then $\chi/Q = \dfrac{1}{F_B \quad R_R}$

F_B = building wake factor

R_R = release rate from reactor building vent (m^3/sec)

Concentration in reactor vent

$C_V = Q/R_R$

where

Q = curies/sec released

Concentration at intake $C_I = C_V/F_B$

C_I also $= Q(\chi/Q)$

Therefore:

$$C_i = \frac{C_V}{F_B} = Q(\chi / Q) = (\frac{Q}{F_B \quad R_R})$$

$$(\chi / Q) = \frac{1}{(F_B)(R_R)} = \text{total dilution factor } (D_F).$$

An F class stability was assumed for atmosphere conditions and 5% meteorology was then applied for time periods from 0 to 180 days. The dilution factors decrease by the following ratios for the time periods indicated.

Time (hr)	0-2	2-8	8-24	24-96	96-4320
Ratio	1.0	0.35	0.04	0.02	0.01

The dilution factors were multiplied by the 5% meteorology ratios to determine the actual χ/Q values used in these computations as presented in Table J.H-1.

J.H.2.2 Integrated Activity Equations Used in this Analysis

The time dependent activity of each nuclide being released from the MSIV was analyzed as follows:

$$\frac{dA1}{dt} = PA_o e\,(-\lambda + \frac{0.0067}{24})\,t \qquad \text{(J.H-3)}$$

where

P = Fractional leak from MSIV per hour (7.1 E-5/hr)

A_o = Initial activity of jth nuclide in primary containment at t = 0 hr

Thus, the activity concentration over a time period of t_1 to t_2 is

$$Q_1 = \int_{t_1}^{t_2} PA_o \; e\left[-(\lambda + 2.8E-4)t\right]$$

or

$$Q_1 = \frac{PA_o}{(\lambda + 2.8E-4)} \left[e^{-(\lambda + 2.8E-4)\ t1} - e^{-(\lambda + 2.8E-4)\ t2} \right] \qquad (J.H-4)$$

The integrated activity concentration from the primary to secondary containment leakage, Q2, was calculated as follows:

$$\frac{dA_2}{dt} = KA_1 - L_2 C_2 - \lambda\ A_2 \qquad (J.H-5)$$

where

K	=	Fractional leak rate from primary containment
	=	$\frac{0.005}{24hr} = 2.1E-4\ hr-1$
A_0	=	Activity in primary containment
	=	$A_o exp\left[-(\lambda + \frac{0.0067}{24})t \right]$
A_1	=	Initial activity (Ci) at t = 0
$\frac{0.0067}{24}$	=	Leakage removal rate from primary containment per hour
	=	2.8E-4 hr^{-1}
L2	=	Discharge rate from reactor building vent via standby gas treatment
	=	2430 ft^3/min (60 min/hr)
	=	1.46E+5 ft^3/hr
C2	=	Activity concentration in secondary containment
A2	=	Curies in secondary containment
V2	=	Volume in secondary containment

Rearranging

$$\frac{dA2}{dt} = kA_o exp\left[-(\lambda + 2.8E-4)t\right] - \frac{L2}{V2} A2 - \lambda A2$$

or (J.H-5A)

$$dA2 = kA_o e^{-F_1 t} - (\frac{L2}{V2} + \lambda)\ A2$$

$$dA2 = \left[kA_o e^{-F_1 t} - F_2 A2\right] dt$$

where

$$F_2 = \lambda + \frac{L2}{V2}$$

$$F_1 = (\lambda + 2.8E-4)$$

$$A2' = kA_1 - F_2 A2$$

$$A2' + F_2 A_2 = r(t)$$

$$r(t) = kA_o e^{-F_1 t} \qquad \text{(J.H-6)}$$

solving

$$A2 = e^{-F_2 t}\left[(\frac{kA_o}{F_2 - F_1})\right] e^{(F_2 - F_1)t} + C$$

$$\text{at } t = o,\ A2 = o \qquad \text{(J.H-7)}$$

$$c = -0.005\, A_o$$

Thus,

$$A_2(t) = 0.005\, A_o e^{-\lambda t} (1 - e^{-(.0042)t})$$

$$Q2 = \frac{L2\, A_2(t)}{V2} \text{ or} \qquad \text{(J.H-8)}$$

$$Q2 = \frac{1.45E+5\ ft^3/hr}{3.5E+6\ ft^3}\left[0.005\, A_o e^{-\lambda j t} (1 - e^{-C2t})\right]$$

where $C2 = 0.042$ (J.H-9)

thus,

$$Q2 = 2.11E\text{-}4\ A_o e^{-\lambda_j t}\ (1 - e^{-C2t})$$

To determine the integrated concentration:

$$Q2(t) = 2.1E\text{-}4 A_o \int_{t_1}^{t_2} \left[e^{-\lambda t} - e^{-(\lambda + C2)t}\right] dt \qquad \text{(J.H-10)}$$

Solving,

$$Q2 = 2.1E\text{-}4 A_o \left[e^{-\lambda t_1} - e^{-\lambda t_2}\right] \frac{(e^{-C2t_1} - e^{-C_2t_2})}{\lambda + C2} \qquad \text{(J.H-11)}$$

The values of Q1 and Q2 are substituted in for each nuclide and each time period. Then using equation (J.H-1), the dose commitment for each nuclide and each time period may be calculated. These results are presented in Section J.6.3.

Table J.H-1

Post-Loss-of-Coolant Accident χ/Q Values[a] Used for Calculations of Integrated Doses Outside the Reactor Building

	Time (hr)				
Area	0-2	2-8	8-24	24-96	96-4320
					(180 days)
Security center	2.1E-4[b]	7.35E-5	8.4E-6	4.2E-6	2.1E-6
Auxiliary security center	1.2E-4	4.2E-5	4.8E-6	2.4E-6	1.2E-6
Sample analysis area (end of cycle)	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Nitrogen supply to accumulators	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Standby service water pump valves	1.2E-4	4.2E-5	4.8E-6	2.4E-6	1.2E-6
Remote shutdown room	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Switchgear room 1	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Switchgear room 2	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Radwaste control room	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Battery racks Direct current battery charger Motor control center	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Three motor control centers/ Three switchgears Direct current battery charger and rack	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Diesel oil tanks	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Solid radwaste control panel	2.6E-4	9.1E-5	1.0E-5	5.0E-6	2.5E-6
Sample of elevated release duct	8.0E-4	2.8E-4	3.2E-5	1.6E-5	8.0E-6

The standby service water pump valves are approximately 700 ft from the release point. This distance is too great to calculate a dilution based solely on a building wake factor. However, the conservative assumption will be made that the dilution at the valves is the same as at the auxiliary guard house which is only 420 ft.

[a] These values are based on an MSIV leak rate of 0.22%/day not the 0.17%/day previously listed. The results are acceptable and conservative for a leak rate of 0.17%/day.

[b] Read as 2.1×10^{-4} etc.